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# ESARDA Symposium 2013
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Proceedings – Part 1

**Edited by**

Filippo Sevini

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Contact information
Filippo Sevini

Address: Joint Research Centre, Via Enrico Fermi 2749, TP 800, 21027 Ispra (VA), Italy
E-mail: filippo.sevini@ec.europa.eu
Tel.: +39 0332 78 6793

http://itu.jrc.ec.europa.eu/
http://www.jrc.ec.europa.eu/

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| Programme | Papers | Participants | Photos |
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The 35th ESARDA symposium on Safeguards and Nuclear Non-Proliferation was held in Bruges, Belgium from 28-30 May, 2013. The Symposium has been preceded by meetings of the ESARDA Working Groups on 27 May 2013. The event has once again been an opportunity for research organisations, safeguards authorities and nuclear plant operators to exchange information on new aspects of international safeguards and non-proliferation, as well as recent developments in nuclear safeguards and non-proliferation related research activities and their implications for the safeguards community.
The Proceedings contain the papers (112) submitted according to deadlines and are available only on-line.

Edited by F. Sevini
European Commission - Joint Research Centre ITU
Ispra, Italy
June 2013


# Programme

Tuesday 28 May, 2013

## Day 1 Morning, 09.00-12.40

**Plenary 1 – <u>Ambassadeur Hall</u>**

9.00: Welcome and opening remarks
*K. van der Meer,* ESARDA President
F. Sevini, ESARDA Secretary

<u>Keynotes:</u>

9.20: *W. Bauwens,* Special Envoy for disarmament, on behalf of the Belgian Minister of Foreign Affairs

09.40: *N. Whiting,* IAEA Director

10.00: Euratom Developments Today and in the Medium Term
*P.Szymanski,* EC ENER E Director

10.20: Horizon 2020 – Views on JRC's R&D in Nuclear Safety, Safeguards and Security
*T. Fanghaenel,* EC JRC ITU Director

**Plenary 2 - <u>Ambassadeur Hall</u>** Chair: K. van der Meer

11.00: INMM Recent Developments
*K. Sorenson*, INMM President

11.20: Implementation of Euratom Safeguards: internal and external state of play (131)
*P. Meylemans, O. Alique, et al*

11.40: Enhanced IAEA/Euratom Cooperation and Reinvigoration of the NPA (126)
*I. Tsvetkov and W.S. Park*

12.00: Japan's Efforts and ISCN's Challenges for Nuclear Nonproliferation and Nuclear Security
*M. Senzaki,* (JAEA)

12.20: What Future Role for ESARDA as a reference technical European Think Tank?
*M. Richard*, ESARDA RG 2010 Chair

**12.40-13.40: Lunch**

## Day 1 Afternoon, 13.40-18.00

| **01 Implementation of Safeguards - <u>Ambassadeur Hall</u>** | **02 Containment & Surveillance - <u>Vives Hall</u>** |
|---|---|
| Chair: R. Dresselaers | Chair: B. Richter |
| 13.40: Experiences with the Development and Implementation of Integrated Safeguards (IS) in some European States (20)<br>*A. Vincze* | 13.40: From DCM14 to the Next Generation Surveillance System – Experience and Plans for Future Implementation (83)<br>*J. Pekkarinen, K. Schoop, et al* |

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|---|---|
| 14.00: The Italian experience in implementing the Additional Protocol (64)<br>*Nadia Cherubini, A. Dodaro, et al*<br><br>14.20: Acquisition Path Analysis Based on Material Flow Directed Graph Methodology (82)<br>*Á. Vincze, A. Lukács, et al*<br><br>14.40: Approaching Acquisition Path Analysis formally – a comparison between AP- and Non-AP States (8)<br>*C. Listner, M. J. Canty, et al*<br><br>15.00: Planning Interim Inspections: The Role of Information (42)<br>*R. Avenhaus* | 14.00: ESAM III – the latest release of Euratom's seal handling application (80)<br>*S. Ciccarello, Peter Schwalbach, et al*<br><br>14.20: Development of an Advanced Ceramic Seal for Maintaining Continuity of Knowledge in Treaty Verification and Safeguards Applications (16)<br>*H. Smartt, J. Romero*<br><br>14.40: Development of a Non-contact Rapid Reader System for Reflective Particle Tags (17)<br>*H. Smartt, M. Sinclair, et al*<br><br>15.00: Enhanced Data Authentication System: Converting Requirements to a Functional Prototype (54)<br>*M. Thomas, G. Baldwin, et al* |
| **15.20-15.40: Coffee break** | |
| **03 Post-Fukushima: NDA Techniques for Molten Fuel Debris - <u>Ambassadeur Hall</u>**<br>Chair: A. Borella<br><br>15.40: Promising NDA Technologies for Material Accountancy of Nuclear Material in Debris of Melted Fuel of Fukushima-Daiichi NPP (113)<br>*M. Seya, H. Harada, et al*<br><br>16.00: Development of neutron resonance densitometry at the GELINA TOF-facility (24)<br>*P. Schillebeeckx, B. Becker, et al*<br><br>16.20: Proposal of Neutron Resonance Densitometry for Particle Like Debris of Melted Fuel using NRTA and NRCA (33)<br>*H. Harada, F. Kitatani, et al*<br><br>16.40: Application of LaBr3 detector for neutron resonance densitometry (56)<br>*H. Tsuchiya, H. Hideo, et al*<br><br>17.00: Particle size inhomogeneity effect on neutron resonance densitometry (22)<br>*B. Becker, H. Harada, et al*<br><br>17.20: Measurement technique for plutonium using nuclear resonance fluorescence with laser Compton scattering gamma-rays (32)<br>*T. Hayakawa, R. Hajima, et al*<br><br>17.40: Improvement of Precision of NRF-Based NDA using Monochromatic Gamma-Ray Beam for Nuclear Materials in Debris of Melted Fuel (31) *T. Shizuma, R. Hajima, et al* | **04 Integrated Measurement and Monitoring Systems – <u>Vives Hall</u>**<br>Chair: W. Kahnmeyer<br><br>15.40: Overview of Next Generation Safeguards Initiative UF6 Cylinder Monitoring Project (13)<br>*K. Durbin, M. Whitaker*<br><br>16.00: A Monte Carlo Analysis of Gas Centrifuge Enrichment Plant Process Load Cell Data (102)<br>*J. Garner, M. Whitaker*<br><br>16.20: Unattended Safeguards Instrumentation at Centrifuge Enrichment Plants (125)<br>*L.E. Smith, A. Lebrun*<br><br>16.40: Modernization of EURATOM's unattended measurement stations at Melox MOX Fuel Fabrication and La Hague Reprocessing Plants (47)<br>*D. Ancius, R. Berndt, et al*<br><br>17.00: The Impact of RDT on Future Safeguards (85)<br>*S. Synetos, Y. Lahogue, et al*<br><br>17.20: Ahaus remote data transmission (RDT) field test – from the operators' point of view (37)<br>*A. Jussofie, K. van Bevern, et al* |

## Day 2 Morning, 09.00-12.40

### 05 NDA I: Neutron Detection
**Ambassadeur Hall**

Chair: P. Peerani

9.00: Capture-gated portable fast neutron spectrometer (65)
*P. Holm, K. Peräjärvi, et al*

09.20: Design of a liquid scintillator-based prototype neutron coincidence counter for Nuclear Safeguards (92)
*A. Tomanin, P. Peerani, et al*

09.40: Modelling of high enriched fission chamber with the code MCNPX (1)
*A. Borella, R. Rossa, et al*

### 06 Destructive Analysis:Quality
**Vives Hall**

Chair: J. Tushingham

9.00: Uranium Isotopic Measurements using a New Uranium Hexafluoride (UF6) Gas Source Mass Spectrometer for Certification of Reference Materials and Nuclear Safeguards Measurements at IRMM (5)
*S. Richter, J. Truyens, et al*

9.20: Investigation on the long-term stability of IRMM-1027 series Large Sized Dried (LSD) spikes (30)
*R. Buják, J. Bauwens, et al*

9.40: Production of monodisperse uranium particles for nuclear safeguards applications (124)
*C. Kim, A. Knott, et al*

10.00: Simple and fast Pu separation for 'fast' Pu screening purposes (123)
*Z. Macsik, C. Kim, et al*

### 07 Information Management I
**Erasmus Hall**

Chair: E. Wolfart

9.00: Mobile Information Technologies and Managed Access during On-site Inspections (10)
*K. Horak*

9.20: New and Emergent Information Technologies for Safeguards (11)
*K. Horak*

9.40: Ontology-based semantic information technology for safeguards: Opportunities and challenges (15)
*M. McDaniel*

10.00: IAEA Safeguards Quality Management System (129)
*D. Korosec, R. McCullough*

10.20: Societal Verification: Intellectual Game or International Game-Changer (78)
*C. Hinderstein, K. Hartigan*

**10.40-11.00: Coffee break**

### 08 NDA II: Neutron Detection
**Ambassadeur Hall**

Chair: A-L. Weber

11.00: Continuous Can Content Monitoring for Special Nuclear Material Control and Accountancy at the Sellafield Product and Residues Store (87)
*L. Bourva, P. Caspell-Askew, et al*

### 09 Destructive Analysis: Measurements – Vives Hall

Chair: Y. Aregbe

11.00: Determination of the incident X-ray source spectrum for the HKED systems used in ITU (7)
*Magdalena Toma, Pieter van Belle, et al*

### 10 Information Management II
**Erasmus Hall**

Chair: K. Horak

11.00: Images Objects vs Pixel: A comparison of new methods from both domains (35)
*S. Nussbaum, I. Niemeyer*

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|---|---|---|
| 11.20: Tracking of fissile material by means of coincident neutron detection - Fission Meter vs. Slab Counter (91)<br>*T. Köble, W. Berky, et al*<br><br>11.40: Expanding the Capabilities of Neutron Multiplicity Measurements: Conclusions from a Four Year Project (28)<br>*B. Goddard, W. Charlton, et al*<br><br>12.00: Fast neutron coincidences from induced fission as a method for detection of SNM (95)<br>*M. Mosconi, A. Ocherashvili, et al* | 11.20: Hybrid K-Edge Densitometry Research at Oak Ridge National Laboratory (104)<br>*R. McElroy, S. Croft, et al*<br><br>11.40: Experiences in Uranium Abundance Analysis Using the MTE Methodology (38)<br>*C. Black, E. Zuleger, et al*<br><br>12.00: Comparative assessment of the Pu content of MOX samples by different techniques (74)<br>*R. Buda, R. Carlos-Marquez, et al* | 11.20: A change detection tool for time series of SAR images (46)<br>*P. Loreaux, G. Quin*<br>11.40: Monitoring uranium mining and processing sites under decommissioning using hyperspectral imagery (63)<br>*I. Niemeyer, C. Listner, et al*<br><br>12.00: Collection And Analysis Of Open Source News For Information Awareness And Early Warning in Nuclear Safeguards (88)<br>*E. Wolfart, G.G.M. Cojazzi, et al*<br><br>12.20: Attaining and Maintaining a Continuity of Knowledge to Draw Safeguards Conclusions with Confidence (117)<br>*R. Bean, D. Blair, et al* |
| **12.40-13.40: Lunch** | | |
| **Day 2 Afternoon, 13.40-17.20** | | |
| **11 Poster Session – <u>Ambassadeur Hall</u>**<br>Chair: F. Sevini<br>13.40 – 14.40<br><br>***Each author will be invited to give a 3 minutes' presentation of the poster, according to the order indicated below. The short presentation could just be oral, or supported by one powerpoint slide which should be provided before the start of the session. Time keeping will be very strict. After the short presentations, the session will continue in the poster area, also during the coffee break.***<br><br>Assessing Capabilities of Portable Raman Spectrometer FirstDefender RM for Complementary Access Applications (121)<br>*A. Berlizov, D. Ho Mer Lin, et al*<br><br>Preliminary NGSI Concept of Operations for a Cylinder Monitoring System (12)<br>*M. Whitaker*<br><br>Application of the two-group - one-region and two-region - one-group Feynman-alpha formulas in safeguards and accelerator-driven system (ADS) (40)<br>*D. Chernikova, I. Pázsit*<br><br>Ultra Compact HPGe Spectrometer For In-Sity Measurements (43)<br>*A. Sokolov, E. Loshevich, et al*<br><br>Sensitivity of the neutronic design of an Accelerator-Driven System (ADS) to the anisotropy of yield of the neutron generator and variation of nuclear data libraries (44)<br>*P. Cartemo, A. Nordlund, et al* | | |

Geology for the country report (59)
*A. Colin, D. Buigues, et al*

Gamma spectrometry as an early and rapid tool in nuclear forensics (60)
*J. Corcho, C. Wirz, et al*

Investigation on spectroscopic capabilities in plastic-based Radiation Portal Monitors (61)
*F. Rosas, P. Peerani, et al*

Categorization of the Main Techniques of Neutron Coincidence/Multiplicity Analysis (81)
*A. Bolind*

Source Representation for a Virtual Reality based Radiation Detection Training (84)
*T. Moltó-Caracena, J.G.M. Gonçalves, et al*

Measurement of Betatron Electron Beam Parameters (118)
*R. Laas, M. Kroening*

Enhancing the Capabilities of the IAEA Safeguards Analytical Services: New laboratories and techniques to meet future analytical challenges (128)
*G. Voigt, R. McGill*

Preparation of Pu particle quality control materials. (122)
*Kim, T. Kitao, et al*

**14.40-15.00: Coffee break**

**12 Fuel Cycle Back-end**
**Vives Hall**

Chair: K. Axell

15.00: Towards effective safeguards implementation in the geological final disposal of spent nuclear fuel (48)
*M. Murtezi, C. Koutsoyannopoulos, et al*

15.20: Safeguards for Disposal Facility for Spent Nuclear Fuel – Construction phase (69)
*M. Moring, M. Hämäläinen, et al*

15.40: Safeguards implementation and status of Posiva's encapsulation plant and geological repository (25)
*M. Lahti*

16.00: Grey sets theory based approach for assessing the vulnerability of safeguarding a geological repository (70)
*C. Turcanu, L. Mkrtchyan, et al*

16.20: Seismic Monitoring of an Underground Repository in Salt - Results of the Measurements at the Gorleben Exploratory Mine (94)
*J. Altmann*

**13 Knowledge Management and Training**
**Erasmus Hall**

Chair: M. Sbaffoni

15.00: What do we mean with knowledge management? (100)
*T. Jonter, M. Marin Ferrer*

15.20: UK National Occupational Standards and National Training Standards for Nuclear material Accountancy and Safeguards (130)
*B. Burrows*

15.40: A Virtual Reality based Safeguards Surveillance Training Tool (86)
*T. Moltó Caracena, J.G.M. Gonçalves*

16.00: Nuclear Safeguards and Non Proliferation Education and Training, initiatives by ESARDA, INMM and JRC (127)
*W.A.M. Janssens, M. Scholz, et al*

16.20: A new facility for NDA-Safeguards Training (89)
*L. Holzleitner, J. Galy, et al*

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| 16.40: The verification and sealing of spent fuel casks for long-term dry storage in the EU - Euratom's approach and implementation experience (62)<br>*A. Kavka, W. Kahnmeyer, et al*<br><br>17.00: Intrinsic fingerprints inspection for identification of dry fuel storage casks (73)<br>*D. Sednev, D. Demyanuk, et al* | 16.40: ORNL Safeguards Laboratory: A Center of Excellence in Nondestructive Assay Research and a Training Center for Internaional Safeguards (107)<br>*A. Lousteau, S. Croft, et al*<br><br>17.00: Laboratory for Nuclear Safeguards, Security and Forensics (116)<br>*J. Bagi, A. Kovács, et al* |
| **19.30: Social Event – bus leaves at Oud St. Jan at 18.45** | |

## Day 3 Morning, 09.00-13.20

### 14 NDA III: Measurements
**Ambassadeur Hall**

Chair: P. Chard

9.00: Determination of Accurate Gamma-Ray Signatures for 233U (4)
*L. Evans, R. McElroy, et al*

9.20: Field Use of In Situ Object Calibration Software (ISOCS) for Gamma Spectroscopy Based Nuclear Safeguards Applications (109)
*A. Bosko, G. Ilie, et al*

9.40: Determination of the half-life and specific thermal power of 241Pu by nuclear calorimetry (120)
*S. Croft, P. Santi, et al*

### 15 Safeguards by Design
**Vives Hall**

Chair: A. Rezniczek

9.00: Proliferation resistance features of reprocessed uranium (112)
*K. Abbas, et al*

9.20: Schematic design and safeguards instrumentation of a Gen IV fuel recycling facility (66)
*M. Åberg Lindell, et al*

9.40: Proliferation resistance considerations within the collaborative project for a European sodium fast reactor (97)
*F. Alim, et al*

10.00: Improved proliferation resistance of fast reactor blankets manufactured from spent nuclear fuel (50)
*C. Hellesen, et al*

### 16 Non-proliferation, Arms Control, Export Control – Erasmus Hall

Chair: I. Niemeyer

9.00: JRC CBRN activities: Contribution to the Implementation of the EU CBRN Action and to the CBRN Centres of Excellence Initiative (29)
*S. Abousahl, Z. Palajova*

9.20: Arms Control Verification Research (71)
*A. Richings*

9.40: Applicability of Nonproliferation Tools and Concepts to Future Arms Control (96)
*M. Dreicer , G. Stein*

10.00: First activities of the new ESARDA Export Control Working Group (3)
*F. Sevini, S. Zero, Q. Michel*

10.20: Managing Threats from Emerging Technologies: Can Safeguards Show the Way? (79)
*T. Leffer*

**10.40-11.00: Coffee break**

### 17 Spent Fuel Verification I
**Ambassadeur Hall**

Chair: A. Borella

11.00: Prototype Development and Field Trials under the Next Generation Safeguards Initiative Spent Fuel Non-Destructive Assay Project (53)
*S. Tobin, T. Tolist*

11.20: A direct method for evaluating the concentration of boric acid in a fuel pool using scintillation detectors (18)

### 18 Safeguards, Safety & Security – Vives Hall

Chair: M. Moring

11.00: 3S by Design for Engineering-scaled Pyroprocessing Facility (2)
*H. Kim*

11.20: Dynamic Fault tree and Event tree assessment for Security system (14)
*A. Sitdikova, M. Čepin, et al*

### 19 Disarmament Verification
**Erasmus Hall**

Chair: M. Richard

11.00: Concepts for dismantlement verification and neutron multiplicity measurements for plutonium mass attribute determination (26)
*M. Göttsche*

11.20: Confirmation of Nuclear Treaty Limited Items: Pre-dismantlement vs. Post-dismantlement (52)
*D. MacArthur, D. Hauck*

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|---|---|---|
| *D. Chernikova, K. Axell, et al*<br>11.40: A laboratory device for developing analysis tools and methods for gamma emission tomography of nuclear fuel (19)<br>*P. Jansson, S. Grape, et al*<br><br>12.00: Recent modelling studies for analysing the partial-defect detection capability of the Digital Cherenkov Viewing Device (58)<br>*S. Grape, S. Jacobsson Svärd, et al*<br><br>12.40: Lessons learned in implementing DCVD partial defects training (67)<br>*D. Parcey, R. Kosierb, et al* | 11.40: SSAC Coordination of Safety, Security, and Safeguards of Nuclear Facilities: A Framework for Analysis (72)<br>*S. Mladineo, S. Frazar, et al*<br><br>12.00: Safeguards-, Safety- and Security-by-Design after Fukushima (103)<br>*J. Pilat, K. Budlong Sylvester* | 11.40: Development of the UK-Norway Information Barrier (75)<br>*K. Allen*<br><br>12.00: Increased transparency in simulations of measurements for nuclear disarmament verification (93)<br>*M. Kütt, M. Englert*<br><br>12.40: The 'Room Within A Room' Concept for Monitored Warhead Dismantlement (114)<br>*J. Tanner, J. Benz, et al*<br><br>13.00: Templating As A Chain Of Custody Tool For Arms Control (115)<br>*J. Benz and J. Tanner* |
| **13.20-14.00: Lunch** | | |
| **Day 3 Afternoon** | | |
| **20 Spent Fuel Verification II**<br>**<u>Vives Hall</u>**<br>Chair: S. Tobin<br><br>14.00: Development of a reference spent fuel library of 17x17 PWR fuel assemblies (6)<br>*R. Rossa, A. Borella, et al*<br><br>14.20: Burnup monitoring of spent fuel assemblies (41)<br>*I. Almási, T. Nguyen, et al*<br><br>14.40: Application of the PDET detector to BWR fuel assemblies: gross defect testing using the spatial distribution of neutron and photon flux (23)<br>*R. Rossa, P. Peerani, et al*<br><br>15.00: Simulations and Preparation for PNAR Measurements of Fugen Fuel (105)<br>*J. Eigenbrodt, W. Charlton, et al* | **21 Panel discussion:**<br>**Disarmament Verification – a Dialogue on Technical and Transparency Issues – <u>Ambassadeur Hall</u>**<br>Chair: G. Neuneck<br>14.00 - 15.40<br><br>Participants:<br><br>• Götz Neuneck (moderator)<br>• Mona Dreicer<br>• David Keir<br>• Ole Reistad<br>• Annette Schaper<br>• Sergey Zykov | |
| **15.40-16.00: Coffee break** | | |
| | | |

| **22 Novel Technologies – Ambassadeur Hall** | **23 Nuclear Forensics – Vives Hall** |
|---|---|
| Chair: H. Toivonen<br><br>16.00: Investing in Next Generation Safeguards Technology Development (98)<br>*T. Sobolev, M. Humphrey, et al*<br><br>16.20: The Analysis and Spectral Assignments of Mixed Actinide Oxide Samples Using Laser-Induced Breakdown Spectroscopy (LIBS) (27)<br>*J.E. Barefield Ii, E.J. Judge, et al*<br><br>16.40: The detection of reactor antineutrinos for reactor core monitoring: an overview (111)<br>*M. Fallot* | Chair: J. Tushingham<br><br>16.00: Development of new analytical capabilities for nuclear forensics at CEA/DIF (21)<br>*F. Pointurier, A. Hubert, et al*<br><br>16.20: Age determination of plutonium for nuclear forensics (39)<br>*M. Sturm, S. Richter, et al*<br><br>16.40: Changes in impurities observed during the refining and conversion of uranium (101)<br>*P. Button* |
| **17.00: Closing plenary**<br>Chair: J. Tushingham | |

Good morning Ladies and Gentlemen, dear ESARDA friends,

For those who do not know me: my name is Klaas van der Meer and as the President of ESARDA it is a real pleasure for me to welcome you at the 35th ESARDA Symposium here in Bruges.

The sign that is shown here behind me may not be used very often during ESARDA symposia. Nevertheless it symbolises the core business of ESARDA and that is peace.

Where lies the real basis of ESARDA? We can consider ESARDA as part of the process to create the European Union, although it has been established later than the fundament of the EU (1969 versus 1957). The early members of ESARDA originated also mainly, but not exclusively, from the 6 founding members of the EU.

Unlike what many believe nowadays, the EU has not been founded for the sake of creating an single economic space to increase the prosperity for the European citizens. The basic idea for the foundation of the EU/EC, as pointed out eloquently by Jan Techau, director of Carnegie Europe, during the 11 November speech last year for the Flemish Peace Institute, was to establish a peace process in Europe in order to erase the root causes for a future war in Europe after the devastating results of the first and second World War. So the EU is not just about a single market, about getting our money back, about the form and size of imported bananas, it is basically about PEACE and establishing a peace process in Europe in order to prevent another war with maybe even more severe consequences than the previous ones. The single market and everything connected to it are just means to obtain that higher goal of peace.

Jan Techau explained during his 11 November speech last year that this peace process is not a given thing. It has been established but it has to be fed, to be nourished in order to maintain it, keep it going and enhance peace in our and other regions.

This peace process has indeed been very successful. It has been so successful that in Europe we consider peace as a given thing, so obvious that it is not worth worrying about. Many think that what happened for instance in ex-Yugoslavia was an incident, nothing more. When I was flying back to Brussels from Vienna a few months ago I was sitting next to a deputy minister of Montenegro on her way to Brussels for the accession negotiations for the EU. She was an economist, but also her main incentive to become member of the EU was to be part of that bigger peace process in order not to have war in her country, rather than wanting money from participating in the EU.

So the European project is basically a peace process. And here ESARDA comes in smoothly, as part of that bigger European peace process.

Talking about peace is nice but I know I have also quite a few engineers in my audience so they are probably wondering when will I get really to the point: what are we going to do in order to support that peace process? So far no real commitments were discussed. So the question is: what is ESARDA doing to contribute in a practical way to this peace process? Well, in our Terms of Reference we state that the scope of ESARDA is to advance and harmonise (=coordinate) safeguards R&D performed by its members.

To do this we have defined 6 purposes for ESARDA, and I simplified them a little for easier understanding:

1. Improve safeguards
2. Consult all stakeholders in order to be proactive
3. Facilitate collaboration
4. Propose new R&D programmes where necessary
5. Find synergies with other verification regimes and technologies
6. Improve communication with public and other experts

In order to achieve these purposes, ESARDA has several activities that can be divided in internal activities (for members only) and external activities, in which other members of the public can participate.

The internal activities, which are not open to the general public, essentially include:

- activities of the permanent technical and scientific Working Groups, our famous ESARDA Working Groups;
- activities of the "ad-hoc" or temporary Working Groups and Committees;
- activities of the management bodies, the maybe less famous Steering and Executive Committee;
- Internal Meetings of the Association, namely the biannual internal meeting/symposium of the ESARDA members in Luxembourg
- activities in bilateral collaboration with other Organisations.

The external activities are open to the general public. They include:

- organisation of Symposia, more particular the biannual open ESARDA symposium;
- organisation of specialised meetings (Seminars, Workshops etc.), like the joint INMM-ESARDA meetings that take place every three years;
- publication of the ESARDA Bulletin;
- web site at the Joint Research Centre.

So the fact that you are sitting here today, participate in this symposium and listen to the various contributions, means that you participate in one of the in fact main activities of ESARDA that have as ultimate purpose the promotion of peace.

What are the current threats and opportunities for this peace process that relate to ESARDA in the sense: where should we as an organisation put our focus on?

Let's start with the bright side of life: what are the opportunities for peace?

First there is the process of nuclear arms reduction and dismantlement. During this symposium various presentations will be given around that theme, both focussed on the more political aspects and on the technical aspects of verification. The safeguards R&D community can contribute with its technical know-how, but will also have to learn to apply verification methods in an environment different from the usual safeguards environment, both in a political sense and a technical sense.

On the global political agenda there is an increased interest observable for non-proliferation in a broader sense, like for instance the Comprehensive Test Ban and the Fissile Material Cut-off. The advantage of these treaties is that the threshold for accession is lower than for a complete nuclear disarmament but that they provide still a strong political signal. The present superpowers have realised or will realise that the presence of a nuclear arms arsenal will in the long term cause more harm in the sense of more horizontal proliferation than it will do good in the sense of providing security to their territory.

On the European level more attention and more financial means are released for non-proliferation in a narrow and broader sense. The development of European instruments for foreign policy, like Instrument for Stability and CBRN Action Plan provide also extra means for our R&D, as will be pointed out for instance by Saïd Abousahl during his speech on Thursday.

Of course these means are not released without the presence of a need to do this. This brings us to the darker side: the threats for non-proliferation!

There is a perceived or real threat with respect to the use of nuclear material by sub-national groups. Nuclear security (often considered together with CBR security) requires partly know-how that is present in the safeguards community and we should be willing to provide this know-how. ESARDA is a good forum to meet partners with expertise in detection methods of nuclear materials, for instance.

Challenges that remain for the more traditional safeguards are the proliferation issues with the DPRK and Iran. While for the DPRK the safeguards world, in particular the IAEA, can only prepare itself for future verification activities when they become possible, Iran is under IAEA safeguards and the situation is reasonably well-known, except the so-called outstanding issues.

Strongly related to these issues is the increased focus on export control in order to prevent sensitive technology to come in the hands of proliferators. ESARDA has an extensive effort in this with the WG on export control, led by our distinguished ESARDA Secretary Filippo Sevini.

Then there remain some challenges that I would characterise as neither bright nor dark.

The implementation of the AP is progressing and has reached a certain maturity in the EU. ESARDA has provided a strong support to the AP implementation via the IS Working Group. Harmonisation of AP implementation throughout the whole EU and in this way an increased efficiency and effectiveness have resulted to the benefit of both the inspectorates and the operators.

The development of new facilities at the back-end of the fuel cycle and in the future GEN IV reactors with possibly new reprocessing techniques asks for new safeguards approaches. The safeguards world should work proactively and maybe a little faster than we did in case of final repositories, where we discussed about safeguards approaches in working groups for more than 15 years and then realised that a repository was already being built without having decided on the approach.
In a similar framework we should also mention the activities of the IAEA with respect to Safeguards-by-Design and support these. Same thing for the State Level Approach and no doubt that Mr. Whiting of the IAEA will say a few words about it.

I personally advocate the importance of having also an eye on the political part of the proliferation equation, since it is the most important one. At the end a decision to develop nuclear weapons or to refrain from them will be a political decision and the technological aspects will only be in the boundary conditions. I am therefore happy to observe contributions from this area and hope for instance that those who will attend the special panel discussion, an initiative by the VTM Working Group under the aeges of Irmgard Niemeyer, will enjoy participating. A cross-fertilisation between the technical and political scientists is necessary for an effective and efficient verification system.

Ladies and Gentlemen, I have almost reached the end of my text here in front of me. I hope that from this speech, you will remember that peace is the core business of ESARDA and I want to ask you that during your stay here in Bruges, at an evening you have dinner or drink one of the fantastic beers, to think about this and make a toast to “world peace”.

With this I would like to open the 35th ESARDA Symposium. I hope you will have a fruitful symposium and I would like to thank you very much for your attention!

# Safeguarding the nuclear future

***Keynote address to ESARDA, 28 May 2013***

## Introduction

Preventing the spread of nuclear weapons is one of the international community's highest priorities. Central to the success of that endeavour is the verification role performed by the IAEA, namely, to ensure that all nuclear material in non-nuclear weapon States remains in peaceful activities and that all nuclear-related activities are solely for peaceful purposes. It is essential that the Agency continues to fulfil this unique role impartially and to the best of its ability.

Today, I want to set out what I regard as the major challenges currently facing the Agency in the area of safeguards and then outline how we are attempting to meet those challenges going forward.

**…**

## The Challenges

The first challenge, I believe, is how to ensure that we continue to implement safeguards effectively.

It is over twenty years since the world was given stark proof that a determined proliferator – in this case Iraq – was able to pursue a clandestine nuclear weapons programme despite having a comprehensive safeguards agreement in force. While the Agency was verifying nuclear material in a declared building, weapons-related activities were taking place in an undeclared building next door, unbeknown to inspectors.

This situation obviously undermined the credibility of the IAEA safeguards system. It triggered efforts to strengthen the Agency's ability to fulfil its obligation under comprehensive safeguards agreements to ensure that safeguards are applied on all nuclear material in the State: in other words, to verify the correctness and completeness of States' declarations. One of the

main outcomes of these strengthening efforts was the approval in 1997 of the Model Additional Protocol, which provided the Agency with additional measures to fulfil that obligation. The implementation of these strengthening measures, particularly those contained in the Model Additional Protocol, resulted in increased access to locations within a State and to information regarding the State's nuclear activities. The Agency also began to exploit newly available technologies, such as environmental sampling; to employ satellite imagery more extensively; and generally to gather, analyse and evaluate a wider range of safeguards-relevant information.

However, the Additional Protocol is voluntary. So, although the 119 States that have one in force today contain the vast majority of States with large nuclear programmes, a small number of such States still remain outside its legal framework. This is important because without an Additional Protocol to complement a CSA, we are unable to conclude that all nuclear material remains in peaceful activities.

A major political challenge, therefore, is to try and achieve universality in adherence to the Additional Protocol. In the meantime, the Agency's major verification challenge is how to provide adequate non-proliferation assurances effectively in those States where only a CSA is in force. Although the Agency has the right and obligation in such States to verify whether their declarations are complete - that is, that all nuclear material has been declared -, in the absence of an Additional Protocol we do not have the necessary measures at our disposal to do so.

This leads me to consideration of our second major challenge, that is, how to deal with the handful of States with which we have encountered serious issues of non-compliance.

Problems with the DPRK date back to 1992 when the Agency was unable to verify the correctness and completeness of that country's initial report on its 'nuclear material subject to safeguards'. The DPRK withdrew from the NPT and subsequently conducted underground nuclear tests. Agency inspectors were never able to verify the DPRK's nuclear programme comprehensively before being ejected from the country in 2009 - and they haven't returned since. Our knowledge of the nuclear programme in the DPRK is, therefore, very limited.

The situation is deeply troubling, and the Director General has called upon the DPRK to to cooperate promptly with the Agency in the full and effective implementation of its NPT Safeguards Agreement. There is little we can do unless and until the DPRK agrees to do so.

In Syria, the existence of what the Agency concluded was very likely to have been a nuclear reactor which should have been declared to the Agency, only came to light after its destruction in 2007. Here too, we have urged Syria to cooperate with the Agency, answer our questions and provide access to locations, people and information. Syria claims that the destroyed building was not a nuclear reactor and the matter remains unresolved.

Iran's clandestine activities only came to light in 2002 with the revelation that, despite its safeguards agreement, it had secretly built an enrichment plant at Natanz and was constructing a research reactor at Arak. Once the Agency was permitted to investigate further, it discovered that Iran had been violating its safeguards agreement for 18 years. Moreover, the Agency has collected and analysed a comprehensive body of information, which overall it assesses to be credible, pointing to the existence of a programme to develop a nuclear payload for a missile. The Agency is now trying to engage Iran to clarify whether there is a possible military dimension to its nuclear programme. But, so far at least, Iran is not cooperating to help us to do so. Moreover, Iran has started to dismantle a site of interest, thereby undermining the Agency's ability to resolve this matter.

Our third major challenge is how to meet rising demand for our services without a commensurate rise in our budget.

Around the world today there is growing interest in nuclear energy, both for electricity generation and other applications. International nuclear cooperation between States is intensifying – with an expansion of trade in nuclear and related equipment, items and materials. If the potential nuclear expansion were to materialize, many additional nuclear activities and facilities would be brought under safeguards.

Of greater concern is the prospect of sensitive nuclear activities, such as enrichment and reprocessing, becoming more widespread. At the same time, growing global trade and the erosion of borders makes it easier for possible

nuclear supply networks to covertly supply nuclear and related technology to interested parties, and the internet facilitates the uncontrolled spread of sensitive nuclear know-how. To summarise: demands on IAEA safeguards are growing and becoming more complex.

In parallel, for the foreseeable future at least, the Agency's budget is not rising.

All of this requires the Agency to rethink how it can invest its resources more efficiently, while ensuring that it does not compromise effectiveness in so doing.

…

## Meeting the Challenges

In my view, these challenges are all interlinked and so are the solutions.

Let me start by stressing the importance of the Agency being able to adapt safeguards implementation to contemporary circumstances.

Since its inception over a half century ago, the Agency and the ways in which it implements safeguards have evolved. When I took over at the Head of the Safeguards Department, building on previous groundwork, I sought to inject impetus to that evolution by enhancing the effectiveness and efficiency of Agency safeguards by further developing and applying a holistic approach that is focussed on the nuclear programme of the State as a whole, the so-called State-level concept.

In essence, the State-level concept, while still acting strictly within our well-established legal mandate, involves the consideration of a State's nuclear programme in the round, rather than as the sum of its declared nuclear facilities. The goal is the more effective and efficient implementation of safeguards.

We are able to adopt this 'State as a whole' approach thanks largely to the availability of more safeguards-relevant information. For instance, we now take and analyse environmental samples, we observe through satellite imagery, we scour huge amounts of open source information, have access to procurement related and third party information, and, if the additional

protocol is in force, we can gain wider access to relevant sites and information. All of this information is now gathered, analysed and assessed within the Department enabling us to conduct a more comprehensive and sophisticated analysis of a State's nuclear programme to help us focus our verification effort.

Under this approach, in determining our verification activities, we take into account a series of objective factors about a State's nuclear programme, which we refer to as "State-specific factors". Nuclear material accountancy remains the bedrock of safeguards and should continue to be so. But it is not the whole picture.

Similarly, although the traditional safeguards criteria still have a role to play in this regard, we are less reliant on them and apply them in a less prescriptive, smarter way. Our aim is to ensure that the full spectrum of safeguards relevant information and increased knowledge about the nuclear programme of a State is fully used in the planning, implementation and evaluation of safeguards activities.

This approach also allows us more easily to explore what room there is for achieving further efficiencies. Do we need to inspect so often, can we make better use of unannounced inspections, can remote monitoring provide a cheaper and equally effective alternative? The implementation of the Additional Protocol is key here too because the measures it contains enable us to build a comprehensive picture of a State's overall nuclear programme and to provide credible assurance that all nuclear material has been declared and that there are no undeclared nuclear activities in the State. It gives us possibilities to further optimise safeguards implementation; to make it more efficient, but also more effective. This is beneficial to both the State and the Agency. Our experience with the successful implementation of integrated safeguards is instructive here, and we are now looking at how we can build upon that and take it to the next level.

In trying to optimise safeguards implementation in a State, the relationship between the Agency and the national safeguards authority can be a critical factor. Building cooperative and trusting relationships brings tangible mutual benefits. In the last two years we have published three new guidance documents aimed at assisting our Member States with the implementation of

safeguards in their countries. This is part of a conscious effort on our part to foster more cooperative partnerships with national safeguards authorities. Real progress is being made, but there is further to go.

Some have argued that, by moving away from a one-size-fits-all approach, the Agency will inevitably begin to discriminate in its implementation of safeguards. But this argument is flawed.

Every Member State is "equal before the law" as determined by their Safeguards Agreement (and possible Additional Protocol thereto), but the nature and extent of their nuclear programme – as well as their record of behaviour, especially their level of cooperation with the Agency - will have a bearing on precisely how we apply safeguards in a State. To allay any fears that this differentiation might become discrimination, we are establishing clear and transparent processes, including those used for State evaluation, for developing individual State-level approaches and for determining the annual verification activities to be undertaken for each State.

We will continue to apply safeguards according to the legal obligations each State has freely entered into. Safeguards implementation will continue to be technically based and safeguards conclusions will remain independent, objective and drawn from the Agency's own findings.

The Agency is currently preparing a report to the Board of Governors - to be issued in time for the September Board meeting - that will provide more details on the State-level concept.

….

Turning to the challenges posed by States with which we have encountered serious issues of non-compliance – what lessons can we learn?

The main lesson would seem to be that, if States are determined to develop an illegal nuclear weapons programme, then their most likely route would appear to be via clandestine activities at undeclared facilities - out of sight of the Agency.

The Agency needs to strengthen its deterrent capability by increasing the chances of non-compliance being detected. If this is where the real

proliferation problem lies, then surely this is where the Agency is duty bound to focus its attention, rather than expending its limited resources over-verifying States just because they have the Additional Protocol in force, have made extensive declarations and provided detailed information to the Agency about their nuclear programme.

To me, the key point here is to preserve the Agency's **credibility** – especially with our Member States. The Agency exercises its authority consistently, robustly and without fear or favour. If we ignore or back down in cases of suspected or actual non-compliance, our authority will drain away and our ability to prevent proliferation will be diminished.

Turning now to how to meet the third challenge: that posed by rising demand for Agency services without a commensurate increase in our budget.

In respect of resources, I have already touched on how we might find greater efficiencies in the ways in which we implement safeguards. In these times of austerity and rising demand for our services, a more strategic targeting of resources is imperative, not just good sense. The Agency depends on the financial contributions of its Member States, many of which are under pressure to reduce public expenditure. We have to spend every Euro wisely. Everything we do has an opportunity cost: we cannot afford to waste money on unnecessary activity. As I said earlier, we need to better focus safeguards implementation on areas of greatest safeguards significance, apply our resources more thoughtfully, focus more on areas of higher safeguards significance, deploy more flexible work practices and exploit new technological solutions.

On the latter point, I believe that the pace of the scientific and technological evolution will offer important opportunities. For instance, computerized devices are evermore capable and innovative technologies, such as artificial intelligence and virtual reality tools, are fast evolving. High-speed digital data networks cover increasingly large portions of the globe. Wireless and satellite communications are more ubiquitous. Information fusion and search tools are ever smarter. Storage capacities continue to increase, making the storage of huge volumes of data possible and less expensive. Such advances have the potential to substantially change the IAEA's technical capabilities and tools.

That is why, in January this year, the Agency published its first Long-Term R&D Plan, setting out the capabilities that the Department of Safeguards needs in order to achieve its strategic objectives and key milestones towards achieving those capabilities. Through an active R&D programme, backed by our Member States, we will be able to continue to invest in the best available scientific safeguards equipment and techniques. In this way we can improve our cost-effectiveness and stay one step ahead of those who may try and evade the safeguards system.

We are already modernizing our laboratory facilities and the supporting infrastructure in Seibersdorf. What we call the ECAS project - *Enhancing Capabilities of the Safeguards Analytical Services* - involves the construction of a new Nuclear Material Laboratory building that will provide a secure, flexible, fit-for-purpose facility in which to conduct analysis of nuclear material samples. Thanks to significant financial support from, amongst others, individual EU Member States and the European Commission, the building is nearing completion.

…

## The Future

So, what of the future?

To succeed in the battle against nuclear weapons proliferation, the world will continue to need the assurances provided by the verification activities undertaken by the IAEA. It will also continue to rely on the legal framework provided by the NPT within which almost all States operate. And the safeguards agreements entered into by virtue of being a party to that treaty will remain the building blocks of the international verification framework.

As long as the nuclear non-proliferation regime remains robust and States are confident that their neighbours can't develop nuclear weapons in secret, we can maintain the regime's integrity. On the other hand, if States were to begin to lose confidence in the regime; if safeguards were no longer seen as effective; if States escaped censure for acts of non-compliance, then the regime would be seriously undermined.

We must persist with efforts to persuade States that have yet to do so, to sign and bring into force an Additional Protocol. The Safeguards Department needs to continue to foster a cooperative relationship with State authorities and facility operators, while sharpening its focus on those few States who fail to cooperate.

As the Director General has emphasized on many occasions, States' adherence to their safeguards agreements with the Agency should remain the benchmark. Of course, the vast majority of States do live up to their safeguards obligations, but where suspicions arise we need to seek clarification, and if non-compliance is uncovered we may need to refer the matter to the Security Council.

My future vision for safeguards is one in which safeguards are implemented in a more focussed way ; in which we draw upon all safeguards-relevant information when conducting our evaluations; in which we continue to draw independent, robust and credible conclusions; and in which non-compliance will be dealt with firmly.

It is vital that we get this right.

The international community demands it.

Future generations depend on it.

And it is our joint responsibility to deliver it.

Thank you for your attention.

European Commission

# Horizon 2020 – Views on JRC's R&D in Nuclear Safety, Safeguards and Security

**Thomas Fanghänel**
**Institute for Transuranium Elements**

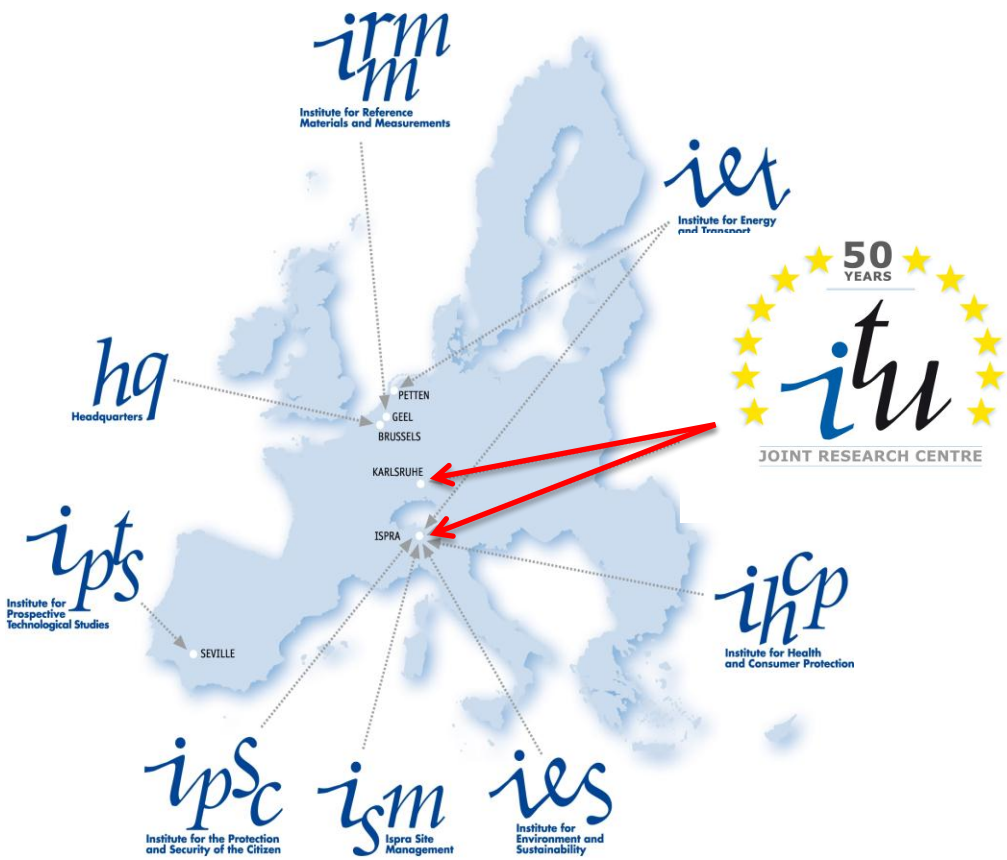


Joint Research Centre


## JRC-Nuclear Safety and Security

Provide independent and reliable scientific and technological assessments on

- Nuclear safety of present and future generations of nuclear reactors,
- Safety of the nuclear fuel cycle,
- Nuclear safeguards, non-proliferation and nuclear security issues.



# Nuclear Security and Non-Proliferation

European Commission

## Nuclear Security and Non-Proliferation: a high international concern


## Reduce risk through a robust, layered defense:

- Eliminating excess stocks of nuclear materials and weapons
- Protecting existing stocks from theft or diversion
- Detecting illicit movement of nuclear or radiological material
- Enhancing nuclear forensics and attribution efforts

Joint Research Centre

# Study 2012


## 2012 interdisciplinary Study (Horizon 2020): Benefits and limitations of nuclear fission for a low carbon economy: defining priorities for Euratom fission research & training

EU Council (meeting of 28 June 2011) requested that the Commission *"organise a symposium in 2013 on the benefits and limitations of nuclear fission for a low carbon economy. The symposium will be prepared by an interdisciplinary study involving, inter alia, experts from the fields of energy, economics and social sciences".*

- 9 scientific-technological experts
- 16 socio-economic experts

⇒ 2013 Symposium *"Nuclear Fission Research for a low carbon economy*", EC and EESC Brussels, 26-27 February 2013

(http://www.eesc.europa.eu/?i=portal.en.events-and-activities-symposium-on-nuclear-fission-forum)

# Study 2012 – synthesis report

## 10 recommendations


available online at http://ec.europa.eu/research/energy/euratom/publications/fission/index_en.htm

# Horizon2020 – Euratom (2014-2018)

European Commission

***Specific objectives for direct actions (fission) JRC as proposed by the commission:***

1. **Improve nuclear safety including: fuel and reactor safety, waste management and decommissioning, and emergency preparedness**
2. **Improve nuclear security including: nuclear safeguards, non-proliferation, combating illicit trafficking, and nuclear forensics**
3. **Raising Excellence in the nuclear science base for standardisation**
4. **Foster knowledge management, education and training**
5. **Support the policy of the Union on nuclear safety and security and the related evolving Union legislation**

Joint Research Centre

# IAEA Long-Term R&D plan:

- Concepts and approaches
- Detection of undeclared nuclear material and activities
- Safeguards methods, equipment and communication
- Information technology, collection, analysis and security
- Analytical services
- New mandates
- Training

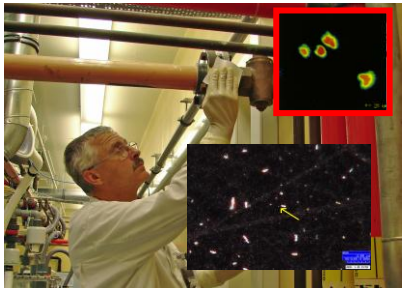

JRC contributions within EC Support programme are well established

# Nuclear Safeguards, Non-Proliferation & Security @ ITU

## Effective and Efficient **Safeguards**

- Nuclear material measurements
- Reference materials
- Containment & Surveillance
- Process monitoring
- On-site laboratories

## Verification of Absence of **Undeclared Activities**

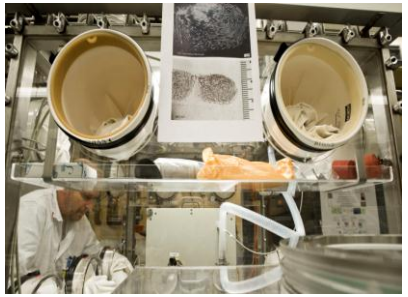


- Trace & particle analysis
- In-field tools for investigative inspector
- Reference materials

## Nuclear **Non Proliferation**


- Export control
- Trade analysis
- Non-proliferation studies

## **Combating Illicit Trafficking**



- Equipment development
- Testing & validation
- Nuclear forensics
- Nuclear preparedness
- National response plan
- CBRN, IfS, …

**TRAINING & EDUCATION** **European Nuclear Safety & Security School**

Promotion /Dissemination of EU's highest Safety/Security standards

# Traditional Safeguards

European Commission

## On-Site Labs at European reprocessing plants

## (support to DG ENER)

**LSS – La Hague**

**OSL - Sellafield**

More than 900 samples per year

Support to Rokkasho

**Refurbishment programme**

- New mass spectrometer for U/Pu analysis at OSL: 2012
- Renovation of infrastructure for NDA measurements: 2013+

Joint Research Centre

# Traditional Safeguards

European Commission

## On-Site Labs at R



M[illegible] per year

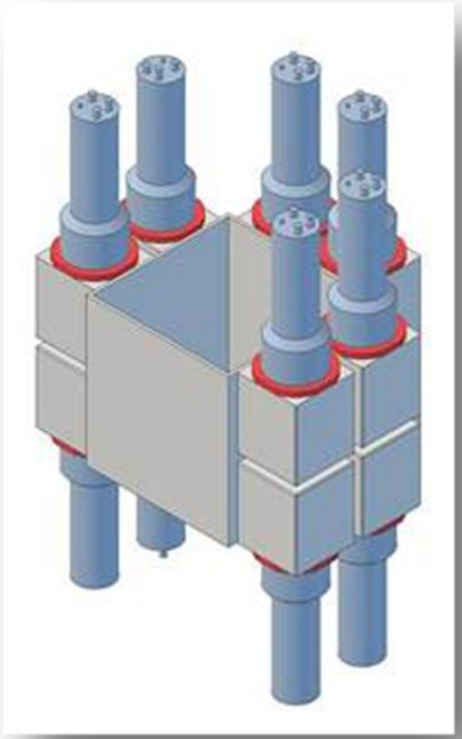
Sup[illegible] to Rokkasho

**[illegible]ovation of infrastructure for NDA measurements: 2013+**

Joint Research Centre

# Testing/design of $^{3}$He-free neutron detectors

**IAEA-SP task:** *conceptual design and prototype testing of a neutron coincidence collar based on liquid scintillators with PSD*



***Detector characteristics:***

- ***12 cubic LS (10x10x10 cm) on 3 sides***
- ***4 mm lead + 1 mm Cd layers***
- ***Cavity able to host PWR/BWR/VVER-440 fuel***
- ***AmLi source in PE on the 4$^{th}$ slab***

***JRC contribution:***

- ***Characterisation of cell performances (at PTB-Braunschweig with IRMM)***
- ***Validation of MCNPX-Polimi model***
- ***Design optimisation/Prediction of performances***

# Traditional Safeguards

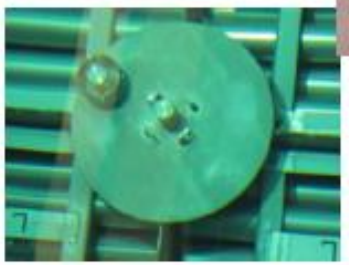
European Commission

## Sealing technology

- Ultrasonic bolt seals for underwater storage : Approved as Category A in 2008.

In 2012 more than 500 seals installed by IAEA and DG ENER throughout the world (Canada-Darlington, France-La Hague, Pakistan-Karachi, Romania-Cernavoda).

A project with ABACC was launched end 2012 for sealing PHWR spent fuel in Argentina-Atucha.

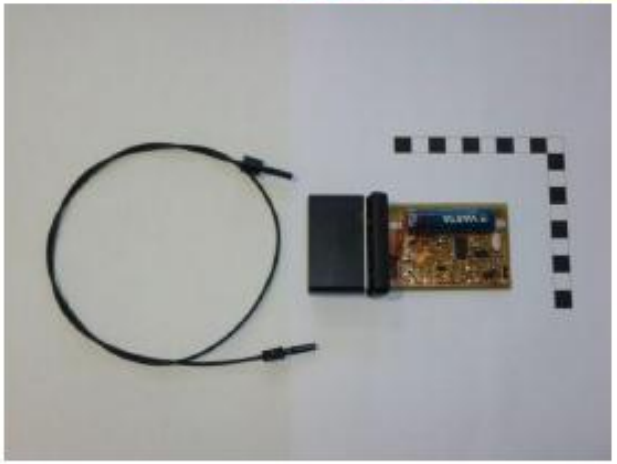


- Low cost electronic seal:

Development of a less than 200 $ seal with active 30 m long optical fiber loop.

Unique authentication and encryption.

Simple user interface and long life batteries.



Joint Research Centre

# Design information verification DIV

European Commission

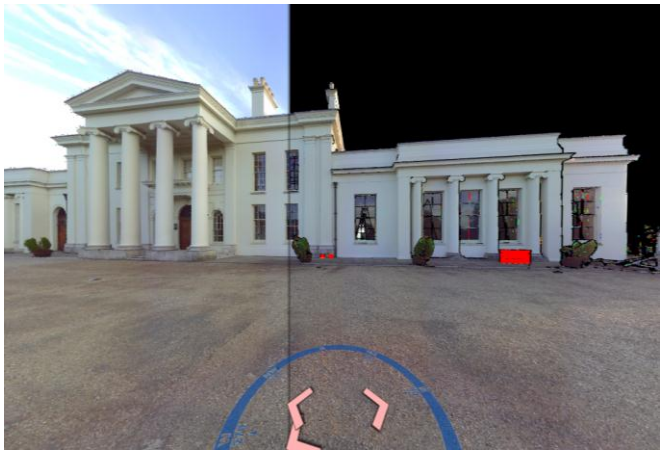
## Integrated Mobile Security Kit (IMSK)

*Implementation, testing and demonstration of:*

- 3DSurveillance system for real-time monitoring
- Innovative 3DViewer to improve the situational awareness.
- 3D Change Detection and Verification

*Final demonstration (Hylands House, UK):*

- Simulating mobile security solution for EU Summit
- Attended by partners and VIP guests
- JRC components were presented in scripted live demo

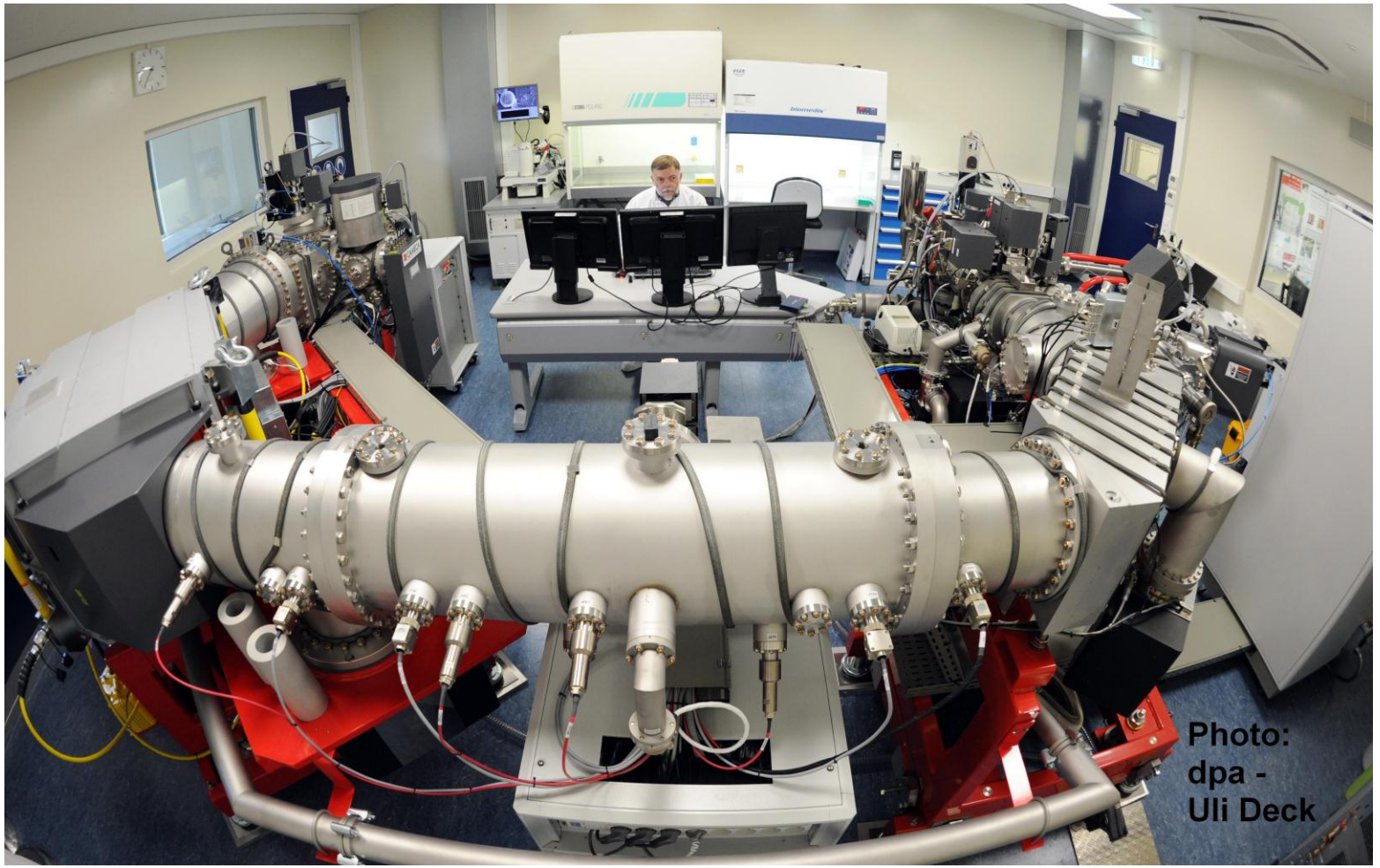


Joint Research Centre

# Strengthened Safeguards

European Commission

# The new ITU LG-SIMS lab




**The new CAMECA IMS 1280HR is now in routine use for Safeguards analysis of trace amount of nuclear materials.**

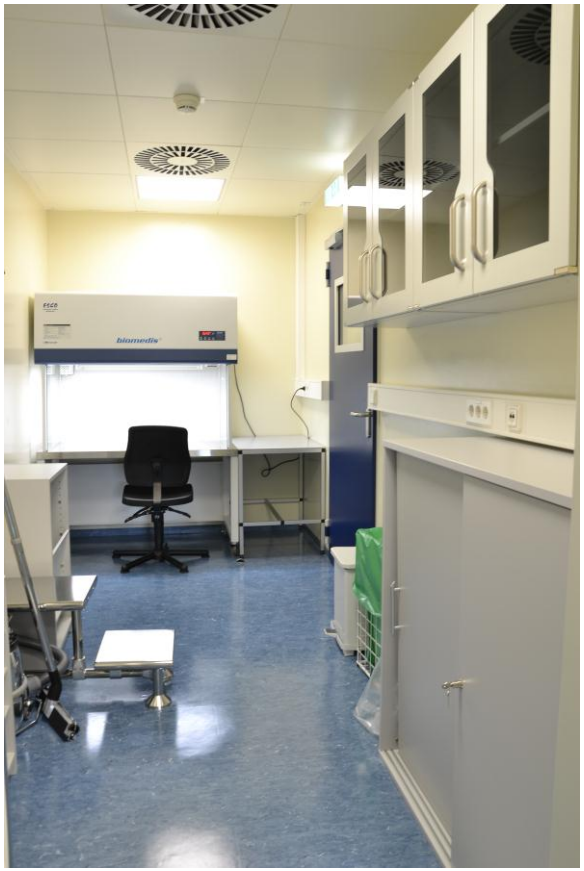
Joint Research Centre

# Strengthened Safeguards

European Commission

# The new ITU LG-SIMS lab

Control room for operating the CAMECA IMS 1280HR and an ASPEX microscope.



Changing and sample receiving area.

Joint Research Centre

**One of the main improvement in uranium particle analysis for Safeguards purposes in the last 10 years is the implementation of LG - SIMS.**

- **The LG - SIMS combines highest quality with speed.**
- **LG-SIMS provides additional information compared to other methods (Distribution of enrichments, elemental information).**
- **The establishment of a LG-SIMS laboratory at ITU is a significant strengthening of the EC's capabilities in environmental sample analysis.**

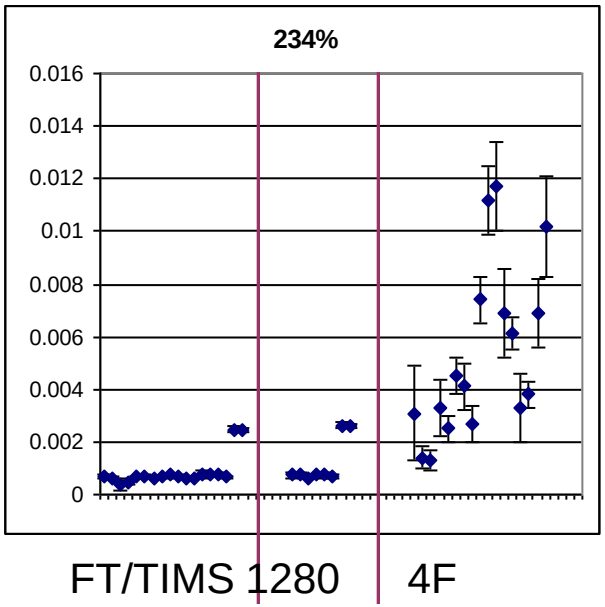
Laboratories that have established LG-SIMS and made instrument time available for Safeguards authorities (IAEA, DG-ENER):

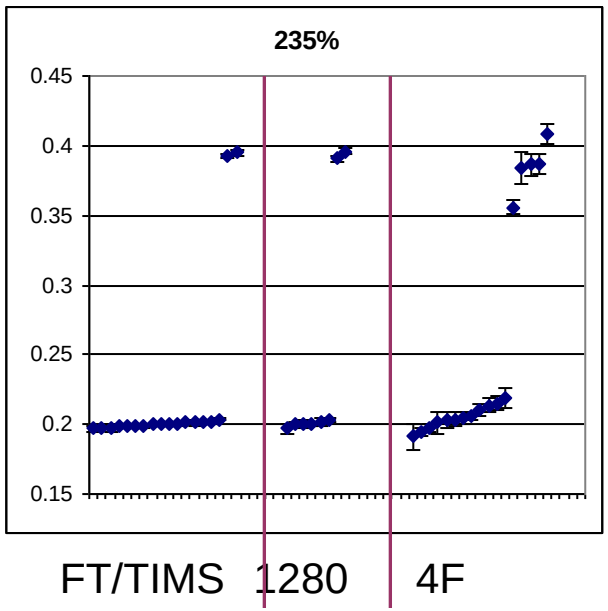
**ESL** (Environmental Sample laboratory), IAEA – Austria
**JRC - ITU**, EC – Germany
**LMA** (Laboratory for Micro Analysis), Russia
**UWA** (University of Western Australia), Perth

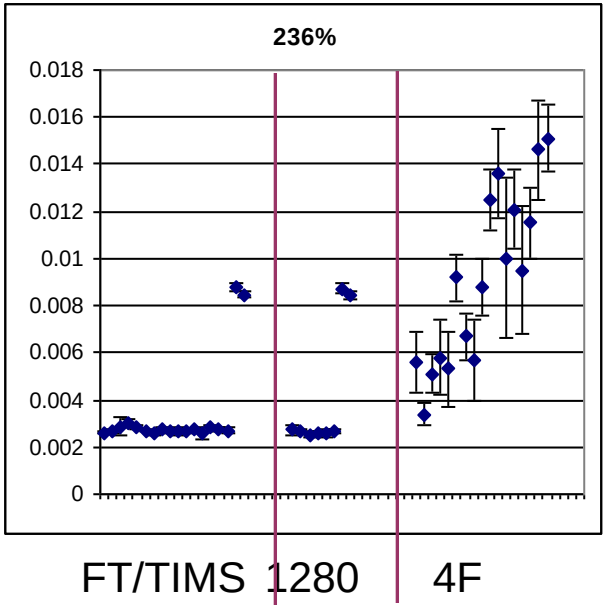
# Strengthened Safeguards

European Commission

# Particle analysis




Comparing FT/TIMS, LG-SIMS (Cameca IMS 1280) and normal SIMS (Cameca IMS 4F). The sample has a high Gd background and is an example where it is very difficult to analyze the minor isotopes with a normal SIMS.

Joint Research Centre

# Nuclear Non-Proliferation

European Commission

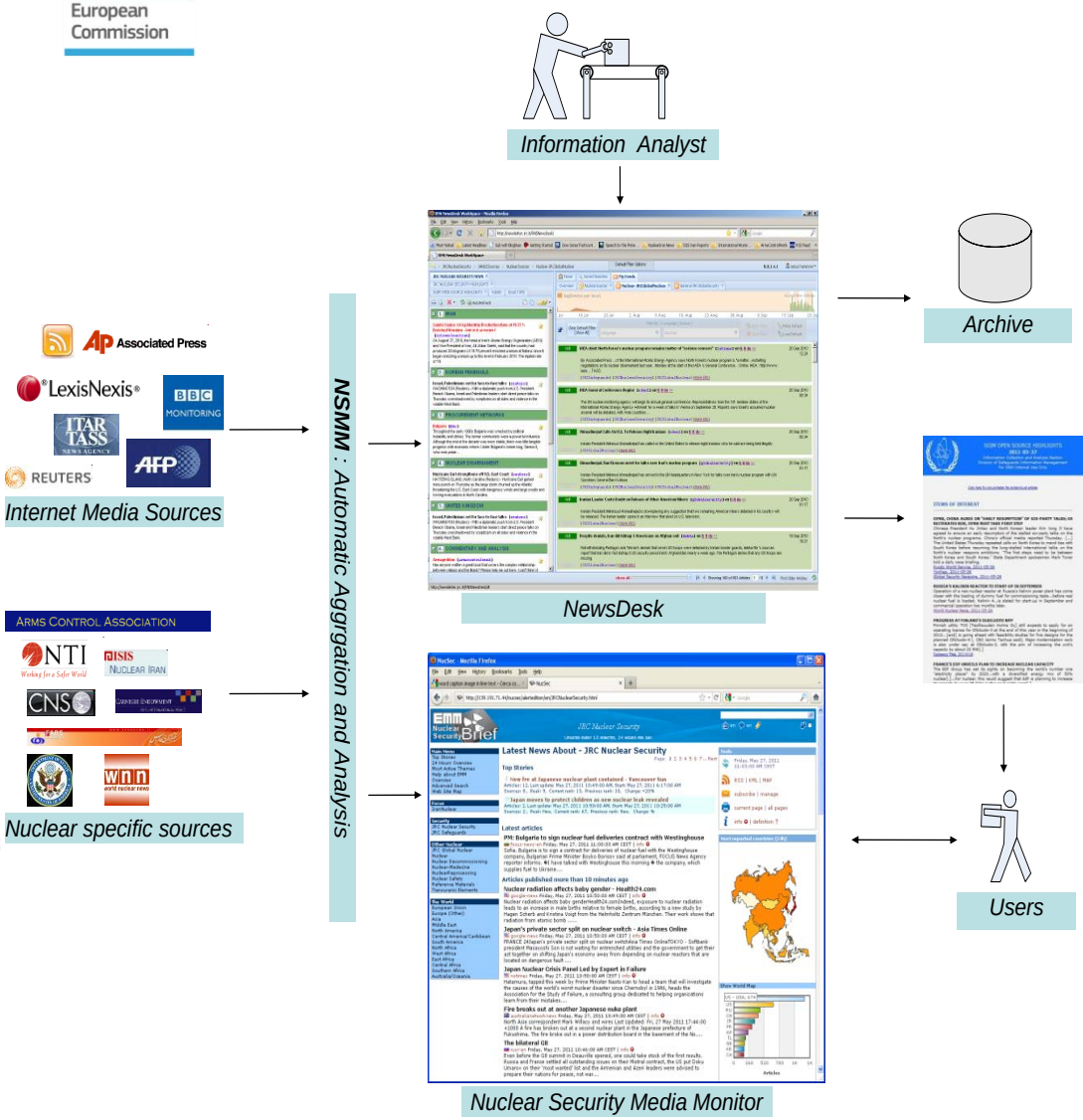
**Trade Analysis:** JRC develops methods & tools

- Software Tool for trade flow analysis
- "The Big Table" (software), now including the IAEA Physical Model

Technical support to **Export control** → new ESARDA Working group since 2012

**Open Source** data mining and analysis

- Collection and dissemination of *JRC Nuclear Security News*
- Information system for nuclear security – *Nuclear Security Media Monitor NSMM*




Joint Research Centre

Information collection and analysis workflow based on NSMM and NewsDesk.

# Combating Illicit Trafficking

European Commission

## Detection

Nuclear Material (U, Pu, reactor or weapons grade) or other radioactive material ($^{60}Co$,$^{137}Cs$, $^{192}Ir$,…)

Detection equipment, intelligence

## Categorization


Joint Research Centre

**20 years of Nuclear Forensics**
1992-2012

## Source Attribution




**EU CBRN Action Plan (2009+):**

- **124 actions to be implemented by EC and EU Member States**
- **Some actions (projects) implemented by EC through JRC**

**On-going projects/actions:**

- **EUSECTRA (Security training Centre) ← Safeguards**
- **Virtual Reality training ← Safeguards**
- **ITRAP+10 (testin gdetection technologies) ← Safeguards**
- **CBRN Glossary**
- **Improvement of IAEA ITDB**
- **Local atmospheric dispersion for CBRN**
  Evaluation of local atmospheric dispersion models and related decision support systems as used by the Competent Authorities in EU
- **Nuclear Forensics ← Safeguards**

# Training & Education @ JRC

European Commission

## EUROPEAN NUCLEAR SAFETY AND SECURITY SCHOOL

- higher academic education (accreditation by academic partners)
- vocational Training
- user facility
- information centre

### Key Areas for EN3S

- Nuclear Safeguards, Security and Forensics
- Nuclear Fuel Cycle, with emphasis on nuclear fuel and fuel processing
- Basic Nuclear Science, with emphasis on the physics and chemistry of the actinides and on nuclear data

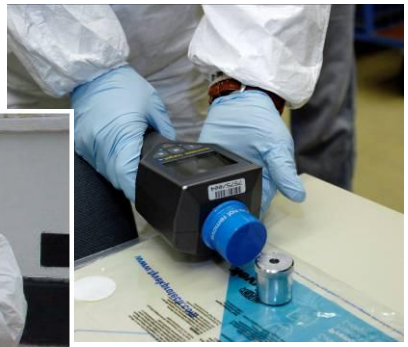
Competences of JRC that are unique or strongly complementary to those of the academic institutions

# Training & Education @ JRC

European Commission

## Ispra & Karlsruhe

**IAEA and DG ENER inspector training**
**- Neutron, Gamma, Mass/volume, DIV, Audit, Statistics**
**- Environmental sampling, COMPUCEA,..**



**Nuclear Forensics & Border Guards Training**

**Virtual Reality Development**

**ESARDA Nuclear Safeguards and Non-Proliferation Course - 10th course in 2013, ENEN recognition (5 day programme)**

**Integrated training for Second Line of Defence Activities**

Joint Research Centre

# EUSECTRA - 18.04.2013: inauguration of KA facilities

European Commission


EUSECTRA

European Nuclear Security Training Centre

# ITU KA New Training areas

**RPM farm**: Training of First Responders (Detection): The focus is on nuclear security training for border guards and Train the trainers.

**Dedicated Security Training area** in the EUSECTRA caisson offering a large collection of equipment and scenario capabilities, including contaminated crime scene management

**Non-Destructive Analysis Training for Safeguards Inspectors** from EURATOM and IAEA. Dedicated floor in the EUSECTRA Caisson training area.

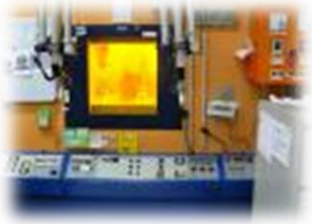
**Highly Specialised Training on Nuclear Forensic Methods**: Mass spectroscopy, Electron Microscopy, Advanced Gamma Spectroscopy, etc. in dedicated laboratories

EUSECTRA

Joint Research Centre

# Additional Protocol Exercise (APEX) at JRC-ITU

- ✓ Pilot project for IAEA in Karlsruhe in Oct 2012
- ✓ Aim: enhance trainees' abilities in using Complementary Access to identify possible indicators and signatures of activities in the area of advanced R&D technologies, especially associated with advanced chemical separation techniques
- ✓ Repeat annually, update and refine scenarios for 2013



Four interconnected scenarios

# International Collaborations

European Commission

50 YEARS itu JOINT RESEARCH CENTRE

ESARDA
European SAfeguards Research and Development Association

IAEA.org
International Atomic Energy Agency

ITWG
Nuclear Forensics

## EURATOM Treaty
**Rome, 1957**

INTERPOL
CONNECTING POLICE FOR A SAFER WORLD

EUROPOL

EUROPEAN EXTERNAL ACTION SERVICE

BMWG
Border Monitoring Working Group

CBRN
Centres of Excellence

Global Initiative to Combat Nuclear Terrorism (GICNT)

GEN IV International Forum

SNETP
Sustainable Nuclear Energy Technology Platform

NUGENIA
NUclear GENeration II & III Association

Joint Research Centre

European Commission

Joint Research Centre

Institute of Nuclear Materials Management

INMM

# ESARDA Symposium 2013
# 35th Annual Meeting

## *INMM:*
## *Current Engagements and Looking to the Future*

Ken B. Sorenson, President
Institute of Nuclear Materials Management

May 28, 2013
Bruges, Belgium

# Current Global Activities

**A few examples…..**

## *2012 Seoul Nuclear Security Summit*

**From the outcome of the Summit, the U.S. committed to:**

- **"…., intending to host a workshop on nuclear security as the chair of the Global Partnership;…"**
  - **This commitment will be fulfilled through an INMM-sponsored workshop entitled: "Risk Informed Security Workshop", October 15-16, 2013, Stone Mountain, Georgia**

- **U.S. committed to "…; intending to support WINS activities."**
  - **Two senior members of INMM sit on the board of WINS**
  - **WINS/INMM Human Reliability Workshop November 2011, Cumbria, U.K.**


# Current Global Activities

**A few examples…..**

***IAEA International Conference on Nuclear Security: Enhancing Global Efforts***

**July 1-5, 2013; Vienna, Austria**

- **INMM will participate as a Cooperating Organization**

***PATRAM2013***

- **INMM Packaging, Transportation, and Disposition Technical Division serves as the principal coordinator for this conference**

***International Safeguard Technical Division Meeting @ 35th ESARDA Symposium***

IAEA
Atoms for Peace
International Atomic Energy Agency
Agence internationale de l'énergie atomique
Международное агентство по атомной энергии
Organismo Internacional de Energía Atómica

Mr Ken Sorenson

President
Institute of Nuclear Materials Management
(INMM)
111 Deer Lake Road, Suite 100
DEERFIELD, IL 60015
UNITED STATES OF AMERICA

Vienna International Centre, PO Box 100, 1400 Vienna, Austria
Phone: (+43 1) 2600 • Fax: (+43 1) 26007
Email: Official.Mail@iaea.org • Internet: http://www.iaea.org

In reply please refer to: CN-203
Dial directly to extension: (+43 1) 2600-21794

2013-03-04

Dear Mr Sorenson,

The International Atomic Energy Agency (IAEA) will hold an **International Conference on Nuclear Security: Enhancing Global Efforts** in Vienna, Austria, from 1 to 5 July 2013. The purpose of the conference is to review the international community's experience and achievements to date in strengthening nuclear security, to enhance understanding of current approaches to nuclear security worldwide and identify trends, and to provide a global forum for policymakers and senior officials to formulate views on the future directions and priorities for nuclear security.

We believe that the conference may be of interest to the Institute of Nuclear Materials Management. Therefore we would like to enquire if your organization is interested in cooperating in this conference.

Cooperating organizations will have their names and emblems on the conference programme, and representatives from such organizations will be encouraged to make presentations during the conference.

We would also appreciate any effort by the Institute of Nuclear Materials Management to disseminate information about this conference.

I would be grateful if we could receive your confirmation that the Institute of Nuclear Materials Management is willing to cooperate in the conference as soon as possible.

Yours sincerely,

Khammar Mrabit
Director
Office of Nuclear Security
Department of Nuclear Safety and Security

# Current Global Activities

## A special word to our ESARDA colleagues….

**Best practices organizations like INMM and ESARDA must be flexible and responsive to the dynamic world of nuclear materials management**

**The evolving linkages between proliferation risks, nuclear programs in non-weapons states, and peaceful use of civilian nuclear power create an environment that requires constant assessment of how we conduct our business**

INMM
INSTITUTE OF NUCLEAR MATERIALS MANAGEMENT

ESARDA
European SAfeguards Research and Development Association

**A close INMM/ESARDA relationship is critical to effectively impact the issues associated with nuclear materials management. We are actively working to strengthen this relationship:**

- **Letter of Intent between INMM/ESARDA, signed Dec. 2011**
- **Cooperate on INMM and ESARDA Working Groups**
- **Strong membership commitment to participate in both INMM and ESARDA activities**

# INMM Snapshots…

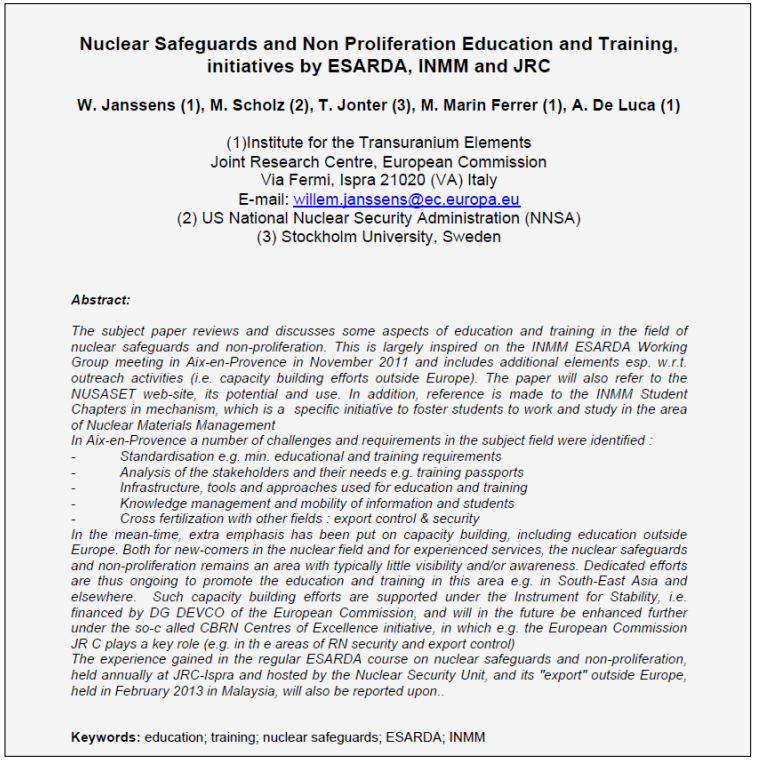
## 1. Leadership transitions

- Jim Larrimore of the International Safeguards Division is retiring!
  Mike Whittaker will assume the ISD Chair position at the INMM Annual Meeting

- Joyce Connery is the Chair of the Nonproliferation and Arms Control Division

- Teressa McKinney is the new Chair of the Technical Program Committee

- Willem Janssens assumed the role of Chair of the Chapter Relations Standing Committee



**Nuclear Safeguards and Non Proliferation Education and Training, initiatives by ESARDA, INMM and JRC**

**W. Janssens (1), M. Scholz (2), T. Jonter (3), M. Marin Ferrer (1), A. De Luca (1)**

(1)Institute for the Transuranium Elements
Joint Research Centre, European Commission
Via Fermi, Ispra 21020 (VA) Italy
E-mail: willem.janssens@ec.europa.eu
(2) US National Nuclear Security Administration (NNSA)
(3) Stockholm University, Sweden

***Abstract:***

*The subject paper reviews and discusses some aspects of education and training in the field of nuclear safeguards and non-proliferation. This is largely inspired on the INMM ESARDA Working Group meeting in Aix-en-Provence in November 2011 and includes additional elements esp. w.r.t. outreach activities (i.e. capacity building efforts outside Europe). The paper will also refer to the NUSASET web-site, its potential and use. In addition, reference is made to the INMM Student Chapters in mechanism, which is a specific initiative to foster students to work and study in the area of Nuclear Materials Management*
*In Aix-en-Provence a number of challenges and requirements in the subject field were identified :*
- *Standardisation e.g. min. educational and training requirements*
- *Analysis of the stakeholders and their needs e.g. training passports*
- *Infrastructure, tools and approaches used for education and training*
- *Knowledge management and mobility of information and students*
- *Cross fertilization with other fields : export control & security*

*In the mean-time, extra emphasis has been put on capacity building, including education outside Europe. Both for new-comers in the nuclear field and for experienced services, the nuclear safeguards and non-proliferation remains an area with typically little visibility and/or awareness. Dedicated efforts are thus ongoing to promote the education and training in this area e.g. in South-East Asia and elsewhere. Such capacity building efforts are supported under the Instrument for Stability, i.e. financed by DG DEVCO of the European Commission, and will in the future be enhanced further under the so-c alled CBRN Centres of Excellence initiative, in which e.g. the European Commission JR C plays a key role (e.g. in th e areas of RN security and export control)*
*The experience gained in the regular ESARDA course on nuclear safeguards and non-proliferation, held annually at JRC-Ispra and hosted by the Nuclear Security Unit, and its "export" outside Europe, held in February 2013 in Malaysia, will also be reported upon..*

**Keywords:** education; training; nuclear safeguards; ESARDA; INMM

# INMM Snapshots…

## Chapters

**U.S. Chapters**
*California*
*Central*
*Northeast*
*Pacific Northwest*
*Southeast*
*Southwest*

**International Chapters**
*Japan*
*Korea*
*Moroccan*
*Nigeria*
*Obninsk*
*Russian Federation*
*United Kingdom*
*Ukraine*
*Urals*
*Vienna*

**Student Chapters**
*Georgia Institute of Technology*
*Idaho State University*
*Jordan University of Science and Technology*
*Mercyhurst College*
*Monterey Institute of International Studies*
*Triangle-areas Universities*
*Pennsylvania State University*
*Texas A&M University*
*University of Michigan*
*University of Missouri*
*University of New Mexico*
*University of Tennessee*
*University of Washington*

## 2. Chapters

- Our chapter activity is strong and growing
- We currently have 29 chapters that represent:
  - 6 regional chapters in the U.S.
  - 10 regional chapters outside the U.S. that span the globe
  - 13 student chapters
- The newest regional chapter is the Nigerian Chapter
- The newest student chapters are the Jordan University of Science and Technology and the Monterey Institute of International Studies
- New petition for a student chapter from the Pandit Deendayal Petroleum University (PDPU)

# INMM Snapshots…

## 3. Preparing for the future

- INMM's strategic planning efforts have been effective in better aligning our focus with a rapidly changing environment:
  - Engaging globally through IAEA, ESARDA, WINS
  - Working through our chapters to extend our reach
  - Establishing a Strategic Planning Subcommittee
    - Identified over 50 organizations with common missions and objectives
    - Initiating scenario planning to assess uncertain future

- Addressing serious administrative constraints on U.S. government support of INMM sponsored events;
  - All U.S. government supported travel to conferences must be pre-approved.
  - INMM is working hard with its DOE sponsors to reduce the impact of this policy on the INMM Annual Conference and workshops

# Conclusions

- **The nuclear materials management environment is very dynamic…filled with uncertainty and opportunity**
- **Global events continue to influence our approach to nuclear materials management**
- **We, in turn, must be flexible and ready to adapt as conditions warrant**
- **A strong, collaborative relationship between INMM and ESARDA is as important as ever**

# Japan's Efforts and ISCN's Challenges for Nuclear Nonproliferation and Nuclear Security


**Director, Masao Senzaki**


*Integrated Support Center for Nuclear Nonproliferation and Nuclear Security (ISCN)*

*Japan Atomic Energy Agency*




**ESARDA Symposium 2013**
**35th Annual Meeting**

**27-30 May 2013, Bruges, Belgium**

# Outline

1. Background of the Establishment of ISCN

2. Three Main Activities and Cooperation with Domestic/Foreign Organizations

3. Capacity Building Assistance through Human Resource Development Including Training and Education

4. Research and Development of Technologies

5. International Collaboration

# 1. Background of the Establishment of ISCN

**Japan's National Statement at the 2010 Nuclear Security Summit : Establishment of an integrated support center for nuclear nonproliferation and nuclear security (ISCN) in the Japan Atomic Energy Agency (JAEA) to contribute to strengthening nuclear security in Asia and other regions**


**On December 27, 2010, ISCN was established in JAEA. It started its full-scale activities in JFY2011.**


**Prime Minister Noda's Speech at the 2012 Nuclear Security Summit**

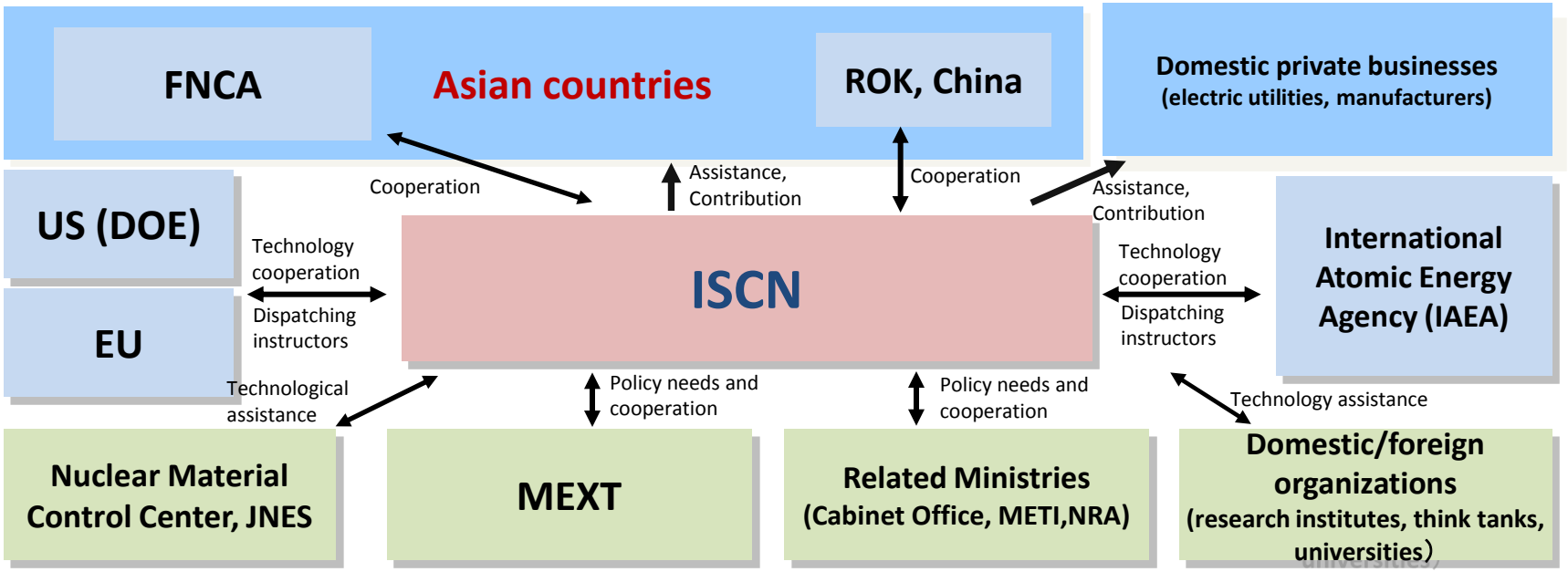
**"In particular, through our 'Integrated Support Center for Nuclear Non-proliferation and Nuclear Security,' established in late 2010, Japan will expand its hosting and training of human resources. "**

# 2. Three Main Activities and Cooperation with Domestic/Foreign Organizations

## Three Main Activities

① Capacity-building assistance through human-resource development, including training and education
② Assistance for infrastructure development
③ Technology development and support for the measurement and detection of nuclear material

## Structure of cooperation with domestic/foreign organizations



# 3. Capacity-Building Assistance through Human-Resource Development, Including Training and Education

## 3-1 Objective

To help ensure that all existing nuclear material is used exclusively for peaceful purposes and is sufficiently protected against theft and sabotage, through:

- Knowledge-sharing,
- Experience-sharing,
- Support for legal development, and
- Hands-on training for state system of accounting for and control of nuclear material (SSAC) and physical protection of nuclear material.

## Needs-Oriented Approach:

### Different Target Participants for Different Programs

- International/Regional Course
- Bilateral Support or Dispatching Course
- Domestic Course

# 3-2 Three Courses for Capacity-Building Assistance

## (1) Nuclear Security Course

① **Target**: Officials/personnel involved in nuclear security policy and regulations, nuclear researchers and operators, radioactive material licensees, etc.

② **Contents**: (a) Design and evaluation process for physical protection systems for nuclear material and facilities, with exercises using the PP training field and the Virtual Reality system, which was introduced in JFY2011
(b) Nuclear forensics, nuclear security culture, INFCIRC/225/Rev.5

③ **Modality**: Two-week course at ISCN, dispatched course at targeted states (2～3 days), workshop (2～3days), domestic course

## (2) SG and SSAC Course

① **Target**: Officials/personnel involved in establishing safeguards and state systems of accounting for and control of nuclear material, nuclear researchers and operators, IAEA inspectors, etc.

② **Contents**: IAEA safeguards, state system of safeguards, material accounting system, safeguards techniques, etc.

③ **Modality**: Two weeks course at ISCN, dispatched course at targeted states (2～3 days), workshop or seminar (2～5 days)

## (3) International Nonproliferation Framework Course

① **Target**: Officials/personnel involved in policy-making or development of domestic legislation in the fields of energy, nuclear power, and nuclear nonproliferation and security; nuclear researchers and operators, etc.

② **Contents**: (a) International trends in the peaceful uses of nuclear energy and in nuclear nonproliferation
(b) International framework and policy in regard to nuclear nonproliferation and nuclear security
(c) Japan's efforts to ensure compatibility between peaceful uses of nuclear energy and nuclear nonproliferation (best practice), etc.

③ **Modality**: Dispatched ISCN staff and distinguished experts to targeted states to hold 2～3 day seminar/workshop

# (1) Nuclear Security Course

**① Regional Training Course on Physical Protection of Nuclear Materials & Facilities:**

Offered for countries in Asia (RTC); Two weeks, including a visit to Hiroshima or Nagasaki

**Cooperation**: U.S. Department of Energy / National Nuclear Security Administration (DOE/NNSA), Sandia National Laboratories (SNL)

**Lecturers**: ISCN, SNL, IAEA, others

**Achievements**: Held twice (2011: 28 participants from 14 countries; 2012: 31 participants from 13 countries); a total of 59 participants attended.

**② Domestic training course:**

Offered for Japanese operators, regulators, and others; Consists of 3 parts of 2-3 days each

**Lecturers**: ISCN

**Cooperation**: DOE/NNSA, SNL

**Achievements**: Held once in 2012; 38 participants attended.

# Nuclear Security Course

## ③ Workshop on INFCIRC/225/Rev.5: Elements of the State's Physical Protection Regime

- **Domestic workshop** offered for Japanese operators and related governmental agencies: 2 days
  **Cooperation**: DOE/NNSA/SNL **Lecturers**: SNL
  **Achievements**: Held once in 2011; 87 participants attended.
- **International workshop** offered for Asian countries: 2 days
  **Cooperation**: DOE/NNSA/SNL **Lecturers**: SNL
  **Achievements**: Held once in 2011; 40 participants from 14 countries (plus Japan) attended.
- **IAEA-JAEA Regional workshop** offered for Asian countries: 2 days
  **Co-hosted** by: IAEA **Cooperation**: DOE/NNSA/SNL
  **Lecturers**: ISCN, SNL, IAEA
  **Achievements**: Held once in 2012; 20 participants from 7 countries (plus Japan) attended.

## ④ Regional Workshop on Nuclear Security Culture

- **IAEA-JAEA regional workshop** offered for Asian countries : 3 days
  **Hosted** by: IAEA **Lecturers**: ISCN, IAEA
  **Achievements**: Held once in 2012; 29 participants from 13 countries (plus Japan) attended.

# Nuclear Security Course

## ⑤ ISCN-WINS workshop

Workshop offered for Japanese operators and related governmental agencies: 1.5 days
**Co-hosted** by: World Institute for Nuclear Security (WINS)
**Achievements**: Held twice in March 2012: 50 participants; September 2012: 63 participants

Theme (Mar.): Nuclear Security and Corporate Governance in Post-3/11 Japan
Theme (Sep.): Collaboration with Outside Organizations for Strengthening Nuclear Security

Theater-based Session : professional actors perform a nuclear-security-event scene based on a specific scenario, followed by discussion among participants on that story.

Feedback from the participants showed that the theater-based sessions were extremely well received. Unlike general lectures or presentations, they provided a more realistic sense of the situations and threats. Also the actors' dramatic performance contributed to a sense of tension. The interest created by the session style led to concrete discussions that greatly benefited the participants.

# Training Field for Physical Protection System

(1) PP System exercise area, (2) Entry Control System exercise area
(3) Fence demonstration area, (4) Performance assessment area, (5) Mock CAS room

Gate

Monitoring and tracking Camera

PP Fence

Monitoring and tracking Camera

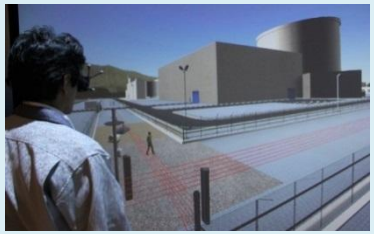
IR sensor

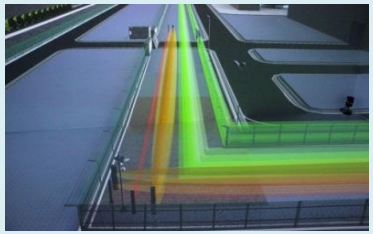
Microwave sensor

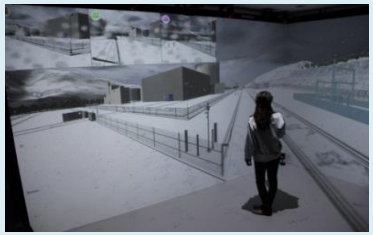
# Virtual Reality System (CAVE System)

- Enhance effectiveness of the training – difficult to conduct nuclear-security training using real nuclear facilities
- Provide a practical training environment for experience-oriented and interactive lessons on nuclear security through a virtual experience of observing the inside/outside of a nuclear facility

**Learning the characteristics of a facility and its physical elements by examining a three-dimensional view of the facility**







**Verify monitoring functions and image features of cameras and sensors**

**Learning skills for handling contingency in a virtual central alarm station**

camera

sensors

metal detection

**Verify installation and functions of security tools**

# Summary - Nuclear Security Course -

| No. | Title of Program (Training Course/Workshop/Seminar) | Domestic/ Bilateral/ International | Venue | Duration | Participants | Remarks |
|---|---|---|---|---|---|---|
| 1 | RTC for Trainer on PP for Nuclear Material and Facilities | Domestic | Japan | 2011/8/22-9/2 | 29 | Japanese only |
| 2 | Workshop on INFCIRC/225/Rev.5 | Domestic | Japan | 2011/9/5-6 | 87 | Japanese only |
| 3 | Seminar on Nuclear Security | Bilateral | Vietnam | 2011-10/13 | 25 | |
| 4 | RTC on PP for Nuclear Material and Facilities | International | Japan | 2011/10/17-28 | 28 | |
| 5 | International Workshop on INFCIRC/225/Rev.5 | International | Japan | 2011/10/31-11/1 | 40 | Japanese : 15<br>USA : 2 |
| 6 | WINS/ISCN workshop on Corporate Governance | Domestic | Japan | 2012/3/5-6 | 50 | Japanese only |
| 7 | PP Inspector Training | Domestic | Japan | 2012-04/21 | 19 | Japanese only |
| 8 | IAEA/JAEA Reginal Training Course on Introduction of Nuclear Forensic | International | Japan | 2012/5/22-24 | 23 | Japanese : 2 |
| 9 | RTC on PP for Nuclear Material and Facilities | Domestic | Japan | 2012/5/30-6/1, 6/27-29, 7/25-27 | 38 | Japanese only |
| 10 | WINS/ISCN workshop on Nuclear Security | Domestic | Japan | 2012/9/4-5 | 63 | Japanese only |
| 11 | Seminar on Nuclear Security | Bilateral | Kazakhstan | 2012/9/25-26 | 18 | |
| 12 | RTC on PP for Nuclear Material and Facilities | International | Japan | 2012/10/15-26 | 31 | |
| 13 | IAEA/JAEA Regional Workshop on INFCIRC/225/Rev.5 | International | Japan | 2012/10/29-30 | 20 | Japanese : 3 |
| 14 | Nuclear Security Training for Chemical School, Japan Groung Self-Defense Force | Domestic | Japan | 2012-11/06 | 21 | Japanese only |
| 15 | IAEA/JAEA Regional Workshop on Nuclear Security Culture | International | Japan | 2012/11/13-15 | 29 | Japanese : 9 |
| 16 | PP Inspector Training | Domestic | Japan | 2012-11/29 | 4 | Japanese only |
| | | | | **Total** | **525** | |

# (2) Safeguards and SSAC Course

## ① International Training Course on State Systems of Accounting for and Control of Nuclear Material

- Training course offered mainly for Asian countries (ITC): 2 weeks (10 days)

  **Cooperation**: IAEA

  **Lecturers**: ISCN, IAEA, US/DOE, Korea/KINAC, Australia/ASNO, EURATOM

  **Achievements**: Held twice and a total of 47 participants attended.

  2011-11/28～12/9： 22 participants from 18 organizations, 12 countries
  Visit to the bombed area: Hiroshima
  Facility visits: Mitsubishi Nuclear Fuel Co., Ltd.; JRR-3 in JAEA

  2012-11/26～12/7： 25 participants from 23 organizations, 18 countries
  Visit to the bombed area: Nagasaki
  Facility visits: Mitsubishi Nuclear Fuel Co., Ltd.; The Japan Atomic Power Company; JRR-3 in JAEA

# Safeguards and SSAC Course

## ② Bilateral Cooperations

- **Vietnam**

**Workshop on the Additional Protocol (AP) Declarations**

Co-hosted by: VARANS (Vietnam Agency for Radiation and Nuclear Safety)
October 11-12, 2012 (Hanoi, Vietnam): 20 participants
July 10-12, 2012 (Dalat, Vietnam): about 30 participants

**Safeguards Capacity Building Training**

To promote capacity building related to safeguards in Vietnam.
February 8-10, 2012 (ISCN, Tokai)
Participants: 9 participants
Sites visited: Tokai Safeguards Center under Nuclear Material Control Center (NMCC), and JAEA's Oarai R&D Center

- **Malaysia**

**Seminar on Additional Protocols**

January 30, 2013 (Dengkil, Malaysia)
To promote the understanding on the importance of the ratification of AP among related governmental officials. Malaysia is about to ratify AP.

**Workshop on Additional Protocol Declarations**

January 31-February 1, 2013 (Dengkil, Malaysia)
To practically train the facility staff who would be responsible for AP declarations.

# Safeguards and SSAC Course

## ③ Training on Safeguards in Reprocessing Plants

- To provide IAEA personnel on-site training at reprocessing plants and for the related R&D

**Lecturers**: JAEA, IAEA

**Achievement**: Mar. 5-9, 2012: 9 IAEA participants and 4 IAEA lecturers
Jan. 28- Feb. 1, 2013: 9 IAEA participants and 3 IAEA lecturers

**Visited Sites**: Tokai Reprocessing Plant (TRP),
Engineering Demonstration Facility-I (EDF-I),
Chemical Processing Facility (CPF) and
Nuclear Fuel Cycle Safety Engineering Research Facility (NUCEF).

## ④ Inspector Training for Spent-Fuel Verification

- To provide the principle, structure, and operation procedure of DCVD (Digital Cherenkov Viewing Device), which is used for spent-fuel verification at the Fukushima Daiichi Power Plant.

**Location**: Hamaoka Nuclear Power Station of Chubu Electric Power Co.

**Lecturers**: From Canada and Sweden as IAEA support states

**Achievement**: June 25-29, 2012: 12 participants
(6 IAEA and 6 NMCC inspectors)

These training courses were held as part of the Japan Support Programme for Agency Safeguards.

# Nuclear Security & SSAC

## ⑤ Participants' Visit to the Atomic Bombed Area (Hiroshima or Nagasaki)

**ISCN has included a visit to a bombed area (Hiroshima or Nagasaki) in its two-week training courses (RTC on Nuclear Security and ITC on SSAC) curriculums. The visit provides participants from other countries with an opportunity to think about the threat of nuclear proliferation as well as to promote better understanding of nuclear nonproliferation and nuclear security.**

### ＜ Participants' Voice ＞

It was my first visit to Hiroshima. In the Hiroshima Peace Memorial Museum, I was impressed with the efforts of how to tell the future generations about the historical experience of the atomic bomb. I felt a strong feeling against nuclear weapons.

"Experience is the best teacher," I felt that. I also felt that we need to make even more efforts (such as nuclear nonproliferation) so that the international community would become one and we could benefit from the peaceful use of nuclear technology.

# Summary - SG & SSAC course -

**SG & SSAC Course**

| No. | Title of Program (Training Course/Workshop/Seminar) | Domestic/ Bilateral/ International | Venue | Duration | Participants | Remarks |
|---|---|---|---|---|---|---|
| 1 | Workshop on Additional Protocol Declaration | Bilateral | Vietnam | 2011/10/11-12 | 16 | |
| 2 | International Training Course on State Systems of Accounting for and Control of Nuclear Material | International | Japan | 2011/11/28-12/9 | 22 | Japanese:2 |
| 3 | Training on SSAC for Vietnamese Inspectors | Bilateral | Japan | 2012/2/8-10 | 9 | |
| 4 | Pilot Training Course on Safeguards in Reprocessing Plants(for Inspectors) | Others | Japan | 2012/3/5-9 | 9 | IAEA only |
| 5 | DCVD Training for Spent Fuel Verification(for Inspectors) | Others | Japan | 2012/6/25-29 | 12 | Japanese:6, IAEA:6 |
| 6 | Workshop on Additional Protocol Declaration | Bilateral | Vietnam | 2012/7/10-12 | 18 | |
| 7 | International Training Course on State Systems of Accounting for and Control of Nuclear Material | International | Japan | 2012/11/26-12/7 | 25 | Japanese:4 |
| 8 | Seminar on Aditional Protocol(AP), Workshop on AP Declaration | Bilateral | Malaysia | 2013/1/30, 1/31-2/1 | 57 | |
| 9 | Safeguarding reprocessing facilities and activities | Others | AUT & JPN | 2013/1/22-24,1/28-2/1 | 9 | |
| | | | | **Total** | **177** | |

# (3) International Nonproliferation Framework Course

## ① Seminar on Peaceful Use of Nuclear Energy and Nuclear Nonproliferation

**【Main contents】**

(1) Peaceful uses of nuclear energy and nuclear nonproliferation/safeguards and nuclear security
(2) Situation of the accident in Fukushima Daiichi Nuclear Power Station and importance of 3S
(3) Functions of governmental organizations, regulatory frameworks of nuclear nonproliferation
(4) Status and prospect of IAEA safeguards

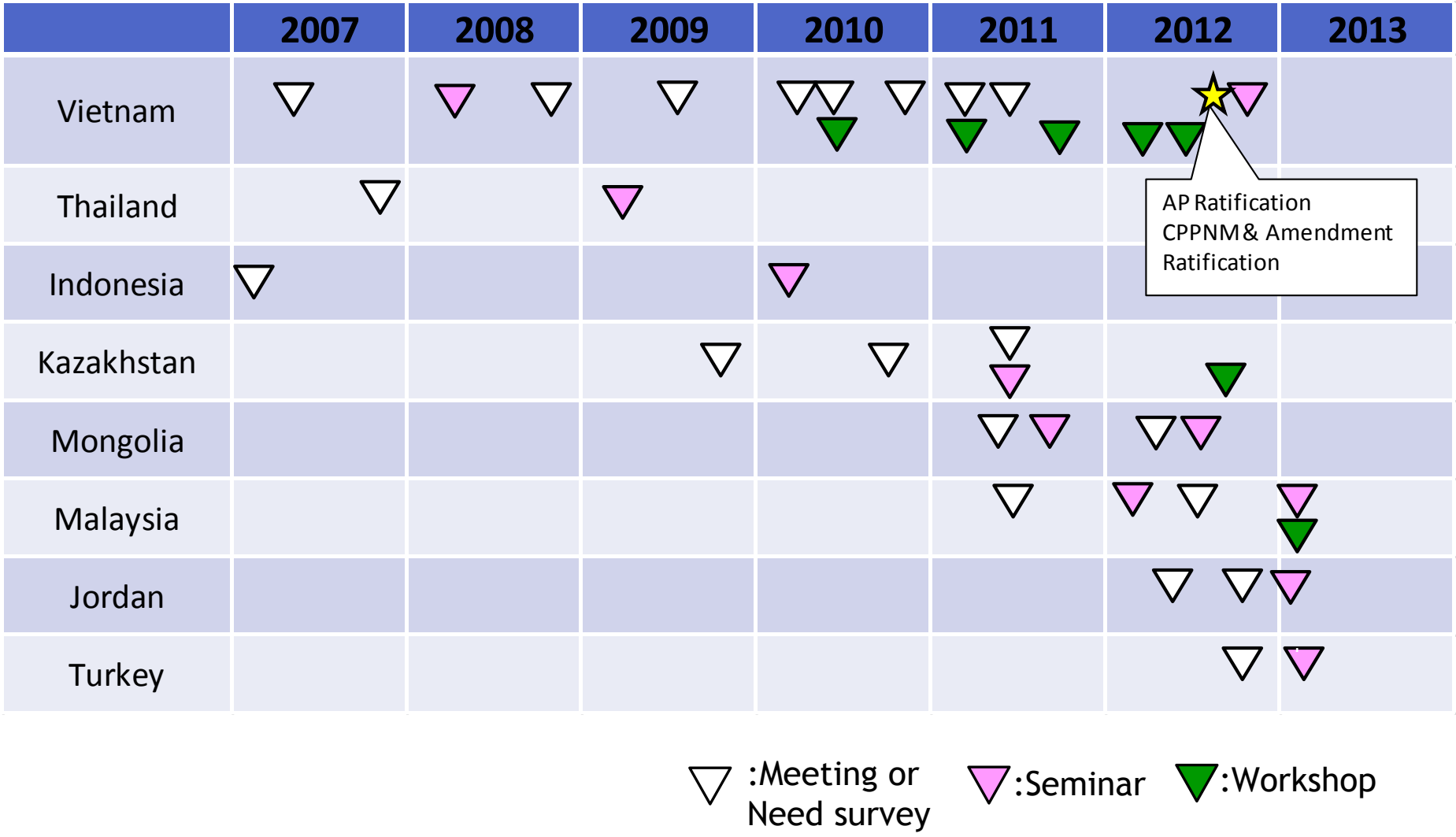
- **Enlighten about the concept of nuclear security**
- **Understand the current situation in the country or the area of the needs of support from ISCN**

## Bilateral Cooperation

**The highest priority is placed on a country…**

- Having nuclear material and
- Planning to introduce a nuclear energy program, but
- Not having an appropriate capacity to ensure its legitimate and secured use of nuclear material.

# ② Bilateral Cooperation Developments with Asian Partners



# Summary - International Nonproliferation Framework Course -

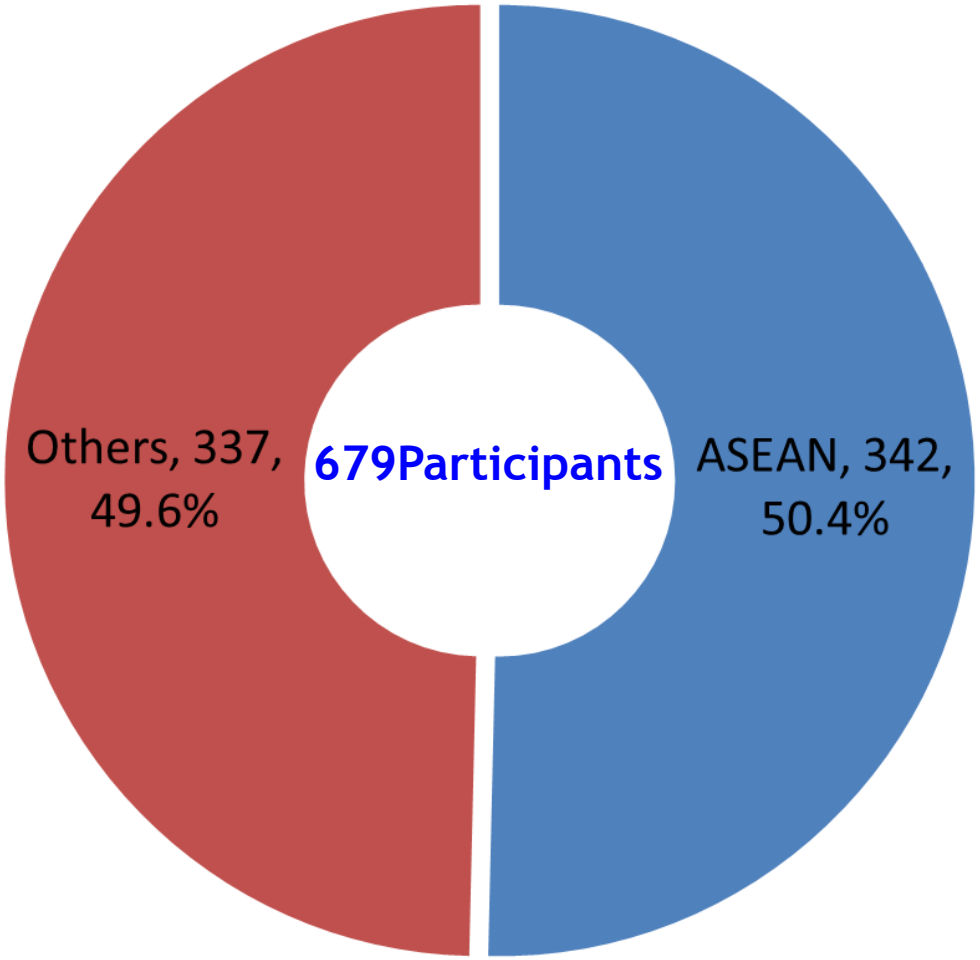
## International Framework Course

| No. | Title of Program (Training Course/Workshop/Seminar) | Domestic/ Bilateral/ International | Venue | Duration | Participants | Remarks |
|---|---|---|---|---|---|---|
| 1 | Seminar on Peaceful Uses of Nuclear Energy and Nuclear Nonproliferation | Bilateral | Kazakhstan | 2011/6/7-8 | 28 | |
| 2 | Seminar on Peaceful Uses of Nuclear Energy and Nuclear Nonproliferation | Bilateral | Mongol | 2011/8/9-10 | 28 | |
| 3 | Seminar on Peaceful Uses of Nuclear Energy and Nuclear Nonproliferation | Bilateral | Malaysia | 2012/2/8-9 | 38 | |
| 4 | Seminar on Nuclear Non proliferation and Nuclear Security for EVN | Bilateral | Japan | 2012/3/13-14 | 10 | for Vietnamese |
| 5 | Seminar on Peaceful uses of Nuclear Energy and Nonproliferation | Bilateral | Mongol | 2012/9/4-5 | 34 | |
| 6 | International Conference of Activities Review and Future Cooperation | Bilateral | Vietnam | 2012-11/20 | 51 | |
| 7 | Seminar on Peaceful uses of Nuclear Energy and Nonproliferation | Bilateral | Jordan | 2013/1/29-30 | 96 | |
| 8 | Seminar on Peaceful uses of Nuclear Energy and Nonproliferation | Bilateral | Turkey | 2013/2/19-20 | 45 | |
| | | | | **Total** | **330** | |

## Others

| No. | Title of Program (Training Course/Workshop/Seminar) | Domestic/ Bilateral/ International | Venue | Duration | Participants | Remarks |
|---|---|---|---|---|---|---|
| 1 | IAEA Consultancy Meeting on Nuclear Forensic | International | Japan | 2012/1/16-19 | 19 | |
| 2 | FNCA Workshop on Nuclear Security and Safeguards Project | International | Japan | 2012/2/22-24 | 27 | Japanese:10 |
| 3 | Physical Protection Exercise for Unversity Student | Domestic | Japan | 2012/9/19-21 | 12 | Japanese only |
| 4 | Physical Protection Exercise for Unversity Student | Domestic | Japan | 2013/3/26-28 | 7 | Japanese only |
| | | | | **Total** | **65** | |

# (4) Proportion of ASEAN Members among All Foreign Participants in ISCN's Courses (JFY2011-2012)




Excluding Participants from Japan and International Organizations

# 4. Research and Development of Technologies

## 4-1 Japan's National Statement at the Washington Nuclear Security Summit (Excerpts)

Ministry of Foreign Affairs of Japan,
http://www.mofa.go.jp/policy/un/disarmament/arms/nuclear_security/2010/national_statement.html

(B) Development of technology related to measurement and detection of nuclear material and nuclear forensics based on international cooperation measurement and detection of nuclear material is a field in which Japan, as an advanced country in the areas of nuclear power, science and technology, should make further contributions….

…Japan will make increased contributions to the international community by establishing these technologies with more precise and accurate capabilities in detection and forensics within an approximate three year time frame and sharing the fruits of these new technologies with the international community.

# 4-2 Measurement and Detection of Nuclear Material (1/3)

JAEA has been conducting research and development on the following technologies of detection and measurement of nuclear material for nuclear security and safeguards:

(1) NRF-NDA technology using laser Compton scattered (LCS) gamma-rays (intense mono-energetic gamma-rays)
(2) NRD using NRTA and NRCA
(3) Alternative to $^{3}He$ neutron-detection technology using $ZnS/B_2O_3$ ceramic scintillator
(4) Spent-fuel measurements with PNAR/SINRD at Fugen

| | | | |
|---|---|---|---|
| NRF | Nuclear Resonance Fluorescence | NRD | Neutron Resonance Densitometry |
| NRTA | Neutron Resonance Transmission Analysis | NRCA | Neutron Resonance Capture Analysis |
| PNAR | Passive Neutron Albedo Reactivity | SINRD | Self-Interrogation Neutron Resonance Densitometry |

# 4-3 Research and Development of Technologies（1/3）

| Developing Technologies | Purpose | For |
|---|---|---|
| (1)<br>NRF-NDA technology using LCS gamma-rays | - Development of the basic technology of high-intensity, mono-energetic gamma-ray production<br>- Basic measurement technology using high-intensity, mono-energetic gamma-rays | - Detection of nuclear material<br>- Precise measurements of nuclear-material isotopes in material that is difficult to measure by conventional NDA |
| (2)<br>NRD using NRTA and NRCA | - Demonstration of the NRD method using a prototype system with a short length | - Detection of nuclear material<br>- Precise measurements of nuclear-material isotopes in a thin, disc-type container |

# Research and Development of Technologies (2/3)

| Developing Technologies | Purpose | For |
|---|---|---|
| (3)<br>Alternative to $^{3}He$ neutron detection technology using $ZnS/B_2O_3$ ceramic scintillator | - Development of the basic technology of $ZnS/B_2O_3$ ceramic scintillator detector<br>- Demonstration of an NDA system using $ZnS/B_2O_3$ ceramic scintillator detectors | - Detection of neutrons<br>- Neutron coincidence measurement of Pu/U (spontaneous fission isotopes by passive measurements, or fissile isotopes by active measurements) |
| (4)<br>Spent-fuel measurements with PNAR/SINRD at Fugen | - Demonstration of an integrated PNAR/SINRD Pu-NDA system | - Measurement of Pu in spent fuel assemblies in a quantitative manner |

# International Collaborations on R&D (3/3)

| R&D | Collaborations with | Remarks |
|---|---|---|
| (1)<br>NRF-NDA technology using LCS gamma-rays | U.S. DOE (LBNL/LANL)<br>- development of simulation codes with benchmarking | Since January 2012 |
| (2)<br>NRD using NRTA and NRCA | JRC (IRMM)<br>- measurement accuracy evaluation | Since May 2012 |
| (3)<br>Alternative to $^{3}He$ neutron detection technology using $ZnS/B_2O_3$ ceramic scintillator | | IAEA has interest in this technology. |
| (4)<br>Spent-fuel measurements with PNAR/SINRD at Fugen | U.S. DOE (LANL)<br>LANL: design/fabrication<br>JAEA: support of measurements | In the course of NGSI<br>Since July 2011 |

In the session "*Post Fukushima: NDA Techniques for Molten Fuel Debris*" of this ESARDA conference, my colleagues present our R&D of (1) and (2).

# 4-4 R&D on Nuclear Forensics

The following three activities will be implemented in cooperation with the U.S. DOE and EC/JRC

➢Uranium Age Dating Measurements(US&EC)

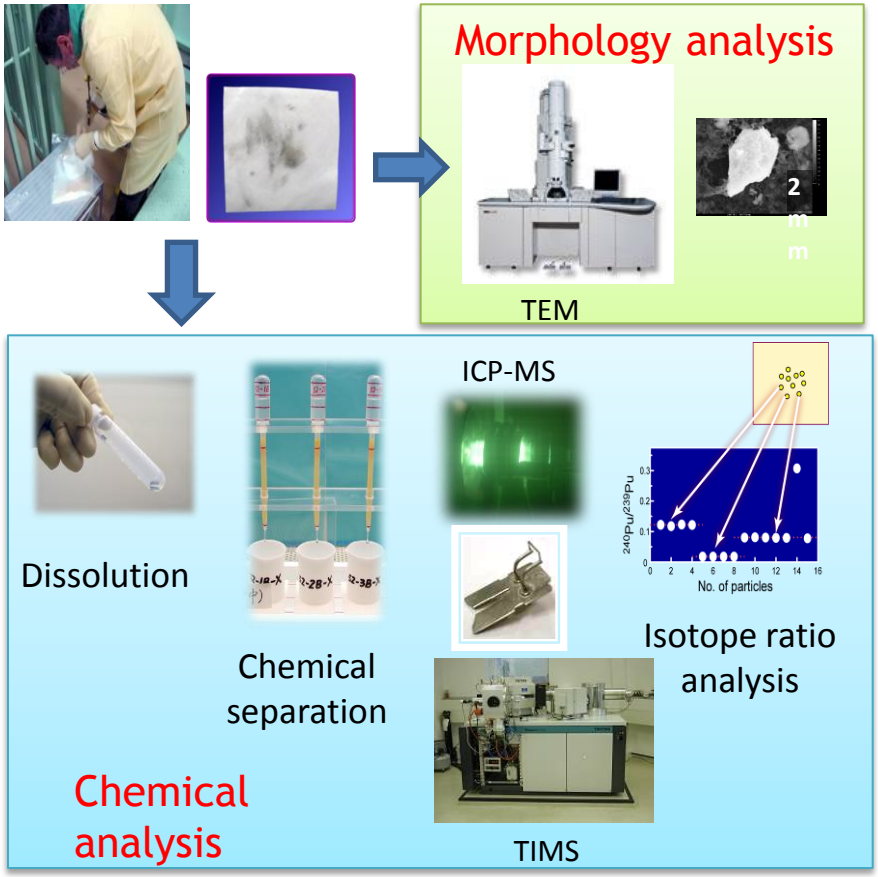
-Procedure exchange
-Procedure review and experimental design
-Age dating inter-laboratory comparison exercise

➢Nuclear Fuel Characterization(US)

-Information exchange for analytical chemistry characterization procedure
-Feasibility study (NM exchange, comparative measurement results, etc.)

➢Nuclear Forensics Libraries(US)

-Study of materials for inclusion into Japanese national nuclear forensics library
-development of a national nuclear-forensics library



# 5. International Collaboration

## < International Organizations >

**IAEA**

- Collaboration on nuclear security and safeguards training

**European Commission (EC) / Joint Research Center (JRC)**

- Human-resource development and technology development for nuclear nonproliferation, nuclear security and safeguards

**World Institute for Nuclear Security (WINS)**

- Workshops on nuclear security

## < Bilateral Cooperation >

**United States: DOE/NNSA, National Laboratories**

- Human-resource development, technology development for nuclear nonproliferation, nuclear security and safeguards
- Agreements for cooperation between JAEA and DOE

**ROK, Vietnam, Malaysia, etc.**

- Memorandum Between the Japan Atomic Energy Agency and Vietnam Agency for Radiation and Nuclear Safety on Cooperation in the field of Infrastructure Development on Safeguards and Nuclear Security for Nuclear Non-proliferation (June, 2010)
- Under discussion with Kazakhstan, Malaysia, etc.
- Under discussion with ROK (KINAC) for agreement for cooperation
  - -Cooperation in sharing and providing information for the Second Nuclear Security Summit

## < Multilateral Cooperation >

**Forum for Nuclear Cooperation in Asia (FNCA)**

- Projects on nuclear security and safeguards from JFY2011; 12 member states
- Exchange of knowledge and information on nuclear security and safeguards, policies and strategies

**Asia-Pacific Safeguards Network (APSN)**

- Organized officially on October 1, 2009; 14 member states and the IAEA
- Improving capabilities of safeguards implementation of pertinent organizations in the Asia-Pacific region

# 5.1 Cooperation between EC/JRC and JAEA (1)

- 28th of May, 1990, Agreement between the Japan Atomic Energy Institute and the European Atomic Energy Community represented by the Commission of the European Communities in the Field of Nuclear Material Safeguards Research and Development (Cooperation, information exchange, researcher exchange)
- 28th of May, 2006, Amendment to the Agreement between the Japan Atomic Energy Agency and the European Atomic Energy Community represented by the European Commission in the Field of Nuclear Material Safeguards Research and Development
- 28th of May, 2011, Amendment to the Agreement between the Japan Atomic Energy Agency and the European Atomic Energy Community represented by the European Commission in the Field of Nuclear Material Safeguards Research and Development

List of specific topics to be studied in the area of cooperation

- Nuclear safeguards and nonproliferation R&D
- R&D in the area of illicit trafficking of radioactive and nuclear materials
- Capacity-building, training, and technical support, in the areas of safeguards, security, and nonproliferation

# Cooperation between EC/JRC and JAEA (2)

## 20th EU-Japan Summit, Brussels, 28 May 2011
## Joint press statement, Annex 6.

- The EU and Japan will strengthen cooperation on mitigation of radiological, nuclear, and other risks in other countries. This includes exchanging information on the implementation of respective programmes, in particular, the EU regional CBRN Centers of Excellence initiative and Japan's Integrated Support Center for Nuclear Nonproliferation and Nuclear Security.

## Recent results of the cooperation between JAEA and JRC in the are of capacity building

- November, 2011: JAEA visited JRC's laboratories for discussion on mutual cooperation.
- February, 2012: JRC dispatched an expert to JAEA's seminar on peaceful uses of nuclear energy and nuclear nonproliferation, in Malaysia.
- March, 2012: JAEA dispatched an expert to the safeguards seminar at ISPRA co-hosted by JRC and ESARDA.
- Dec., 2012:JRC dispatched an expert to JAEA's International Training Course on SSAC
- Jan., 2013:JAEA invited observers from JRC to J AEC/JNRC Seminar on Peaceful Use of Nuclear Energy and Nuclear Nonproliferation
- Feb., 2013: JAEA dispatched experts to the 11th ESARDA Course on Nuclear Safeguards and Non Proliferation in Kuala Lumpur, Malaysia
- March , 2013: JAEA dispatched experts to the 10th ESARDA Course on Nuclear Safeguards and Non Proliferation
- April , 2013: JAEA participated Opening Ceremony of the European Nuclear Security Training Centre (EUSECTRA)

# 5-2 Japan-US Cooperation on Nuclear Nonproliferation and Nuclear Security (1)

President Obama's speech in Prague, April 2009

- In November 2009, Japan and the U.S. issued a joint statement toward the nuclear-free world, agreeing on expanding cooperation in the field of nuclear nonproliferation, safeguards, and nuclear security.
- At the Nuclear Security Summit in Washington, D.C., April 2010, the Ministry of Education, Culture, Sports, Science and Technology (MEXT) and the U.S. DOE signed a Memorandum of Cooperation on nuclear nonproliferation, safeguards, and nuclear security.

**Fields of Cooperation:**

- Development of nuclear measurement and detection technologies
- Coordinate the two countries' programs on technical support to IAEA
- Support the capacity building of emerging nuclear countries through assistance for developing safeguards implementation systems and human resources in the field of nuclear security
- Development of technical measurements for the physical protection of nuclear materials in a nuclear facility

- Japan-U.S. Summit Meeting in November 2010

⇒Establishment of Japan-U.S. Nuclear Security Working Group (NSWG) toward Seoul Nuclear Security Summit in 2012

# Japan-US Cooperation (2)
## Japan-US Nuclear Security Working Group (NSWG)

- Identify and coordinate tangible outcomes for the 2012 Nuclear Security Summit, by promoting robust security of nuclear materials at civilian nuclear facilities and during transportation

- Conduct joint activities in the fields of nuclear forensics, nuclear material detection, and measurement, and in the strengthening of expertise on nuclear security in the Asia-Pacific region where nuclear power utilization is expected to increase.

- Agreed on the importance of Japan's decision to establish an Integrated Support Center for Nuclear Nonproliferation and Nuclear Security.

- This Group has successfully fulfilled its responsibility to identify and coordinate tangible outcomes for the 2012 Nuclear Security Summit, including the promotion of robust security for nuclear materials at civilian nuclear facilities and during transport, by making achievements in the following 9 areas:

**Goal 1:** Co-operation within the Integrated Support Center for Nuclear Non-proliferation and Nuclear Security (ISCN)
**Goal 2:** Research and Development of Nuclear Forensics, Measurement and Detection Technologies, and Sharing of Investigatory Best Practices
**Goal 3:** Cooperation on Safeguards Implementation
**Goal 4:** Sharing Best Practices for Nuclear Security in New Facility Design
**Goal 5:** Cooperation on Transport Security to Reduce the Chances of Theft or Sabotage
**Goal 6:** Convert Reactors to Reduce the Use of HEU and Complete Down-Blending Operations
**Goal 7:** Implement INFCIRC / 225 / Rev.5
**Goal 8:** Integrating Response Forces into Dealing with Theft and Sabotage at Facilities
**Goal 9:** Joint Study on Management of HEU and Plutonium: Reduction of Material Attractiveness


US members visited JAEA, and the second NSWG meeting was held on February 6, 2012.

# ISCN Workshop

## in Conjunction with the IAEA International Conference on Nuclear Security: Enhancing Global Efforts (1-5 July 2013)

### Venue and Schedule

- Organizers: ISCN/JAEA, PMJ（Permanent Mission of Japan to the International Organizations）, VCDNP（Vienna Center for Disarmament and Non-Proliferation ）
- Venue: Main Conference Room in PMJ (24th floor, Andromeda Tower)
- **Time and Date: 13:00 – 14:30, Wednesday, 3 July 2013**（The workshop participants will be invited to lunch ）

### Panel Theme and Contents

- Theme: “Nuclear Security Centers of Excellence in Asia: Harmonization and Nexus”
- Opening remarks: Ambassador Ozawa, PMJ
- Keynote speech: Ambassador Shahrul Ikram bin Yaakob, Malaysia
- Possible panel discussion moderator and panelists

**Moderator**: Ms. Elena Sokova , Executive Director, VCDNP

**Panelists :**

ROK: Dr. Kwan-Kyoo Choe, International Nuclear Nonproliferation and Security Academy (INSA), KINAC

China: State Nuclear Security Technology Center or CAEA

Japan: Mr. Masao Senzaki, Director, ISCN

**Commentators**

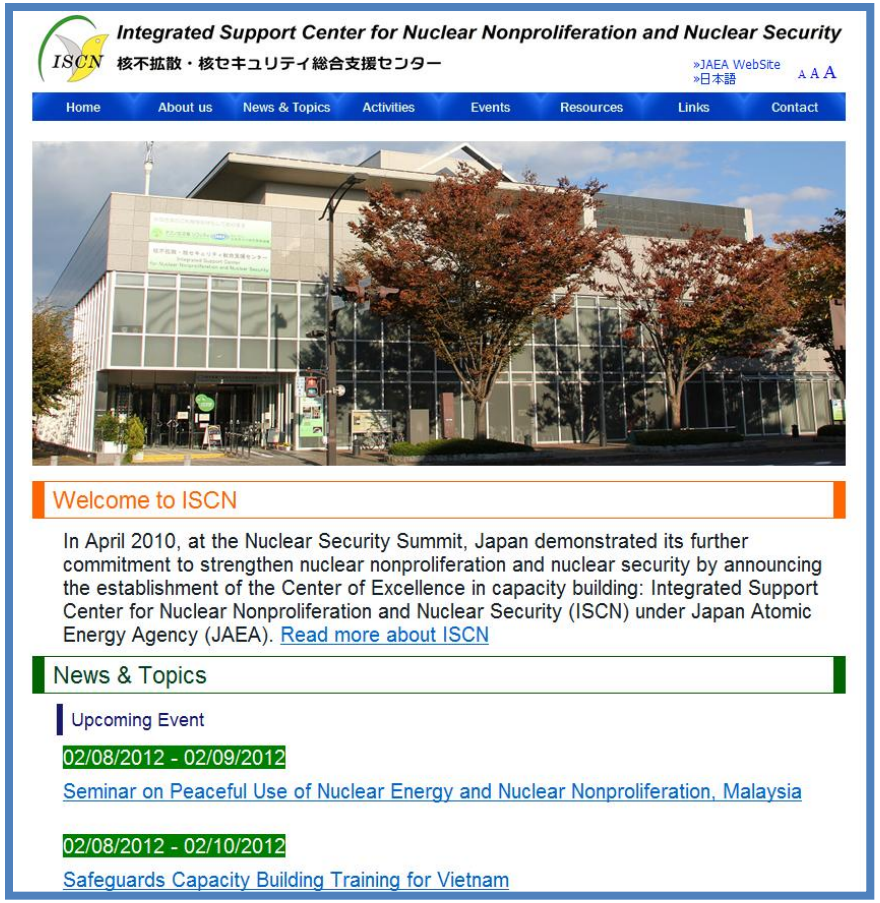
US: Ms. Martie Larson, International Nuclear Security Program Leader, DOE/NNSA

EC: Mr. Adriaan Van der Meer, Head of Unit, EC/DEVCO

IAEA: Khammar Mrabit, Head of the Safety and Security Coordination Section, Department of Nuclear Safety and Security

# Thank you for your attention

**Please visit our website!** http://www.jaea.go.jp/04/iscn/index_en.html



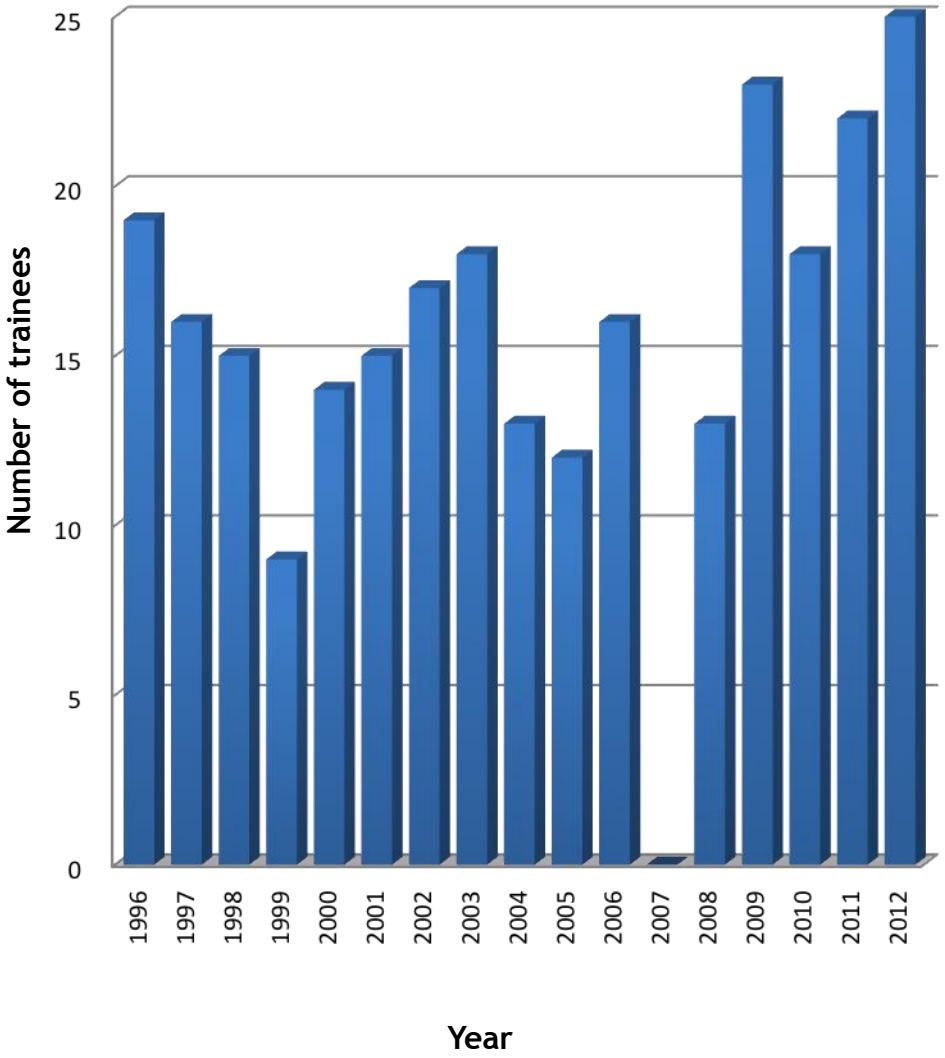

Integrated Support Center
for Nuclear Nonproliferation
and Nuclear Security
Japan Atomic Energy Agency

3-1-1 Funaishikawa Eki Higashi,
Tokai-mura, Naka-gun, Ibaraki, 319-1118 JAPAN
Tel/Fax: +81-29-283-4115

***Integrated Support Center for Nuclear Nonproliferation and Nuclear Security***

# Safeguards/SSAC Course, JAEA/ISCN Experience

**266 participants from 36 countries in total, between 1996 to 2012 (1996-2010: JAEA; 2011-today: ISCN)**




| Country | # of Trainee |
|---|---|
| Armenia | 10 |
| Australia | 4 + 1 |
| Azerbaijan | 1 |
| Bangladesh | 5 + 1 |
| Belarus | 8 |
| Bulgaria | 5 |
| Cambodia | 1 + 1 |
| China | 19 |
| Czech | 4 |
| Estonia | 1 |
| Georgia | 1 |
| Hungary | 2 |
| Indonesia | 22 + 2 |
| Japan | 15 + 4 |
| Jordan | 1 |
| Kazakhstan | 12 + 1 |
| Korea | 13 + 1 |
| Kyrgyz | 1 |
| Lao | 1 + 1 |

(Red: 2012)

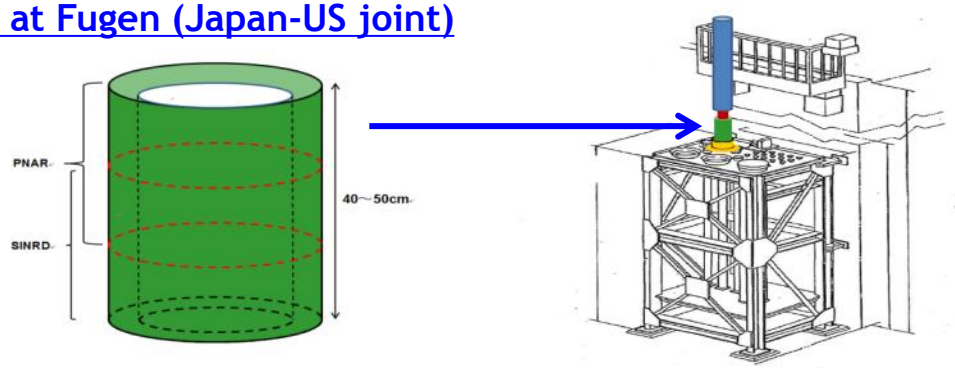
| Country | # of Trainee |
|---|---|
| Latvia | 3 |
| Lithuania | 5 + 1 |
| Malaysia | 16 |
| Moldova | 1 |
| Mongolia | 4 |
| Myanmar | 3 + 1 |
| Nepal | 1 |
| Philippines | 6 |
| Romania | 4 |
| Russia | 15 |
| Singapore | 1 |
| Slovak | 4 |
| Tajikistan | 1 |
| Thailand | 15 + 2 |
| Turkey | 1 + 1 |
| UAE | 1 + 1 |
| Ukraine | 15 + 1 |
| Uzbekistan | 5 |
| Vietnam | 17 + 3 |
| **TOTAL** | **266** |

# Measurement and Detection of Nuclear Material (2/3)

## Spent fuel measurements with PNAR/SINRD at Fugen (Japan-US joint)

**Expected outcome**
- Accurate measurement of the amount of Pu in spent fuel

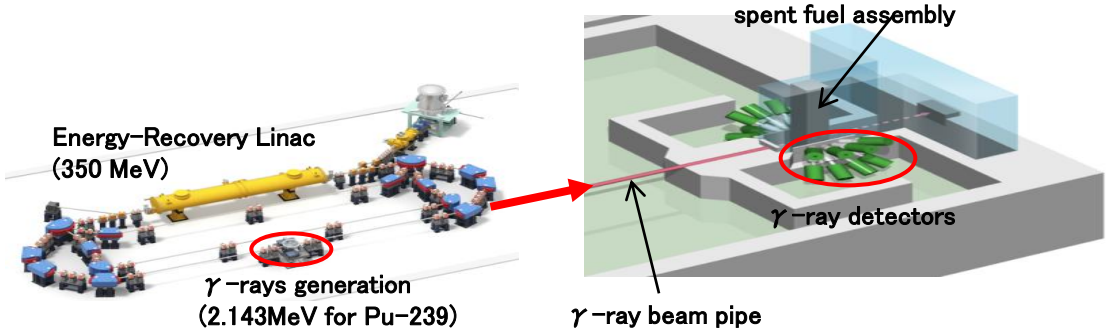
**Time of completion**
- During JFY2013




## NRF-NDA technology using laser Compton-scattered (LCS) gamma-rays (Japan-US joint)

**Expected outcome**
- Detection of fuel-pin removal in spent fuel, realization of reliable detection of nuclear material that is covered with shielding
- Possible application of accounting for and control of nuclear material for melted fuel at Fukushima Daiichi Nuclear Power Station

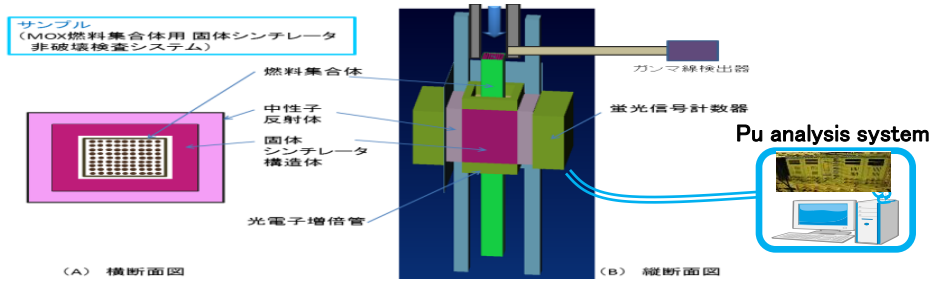
**Time of completion**
- During JFY2014 (basic demonstration)



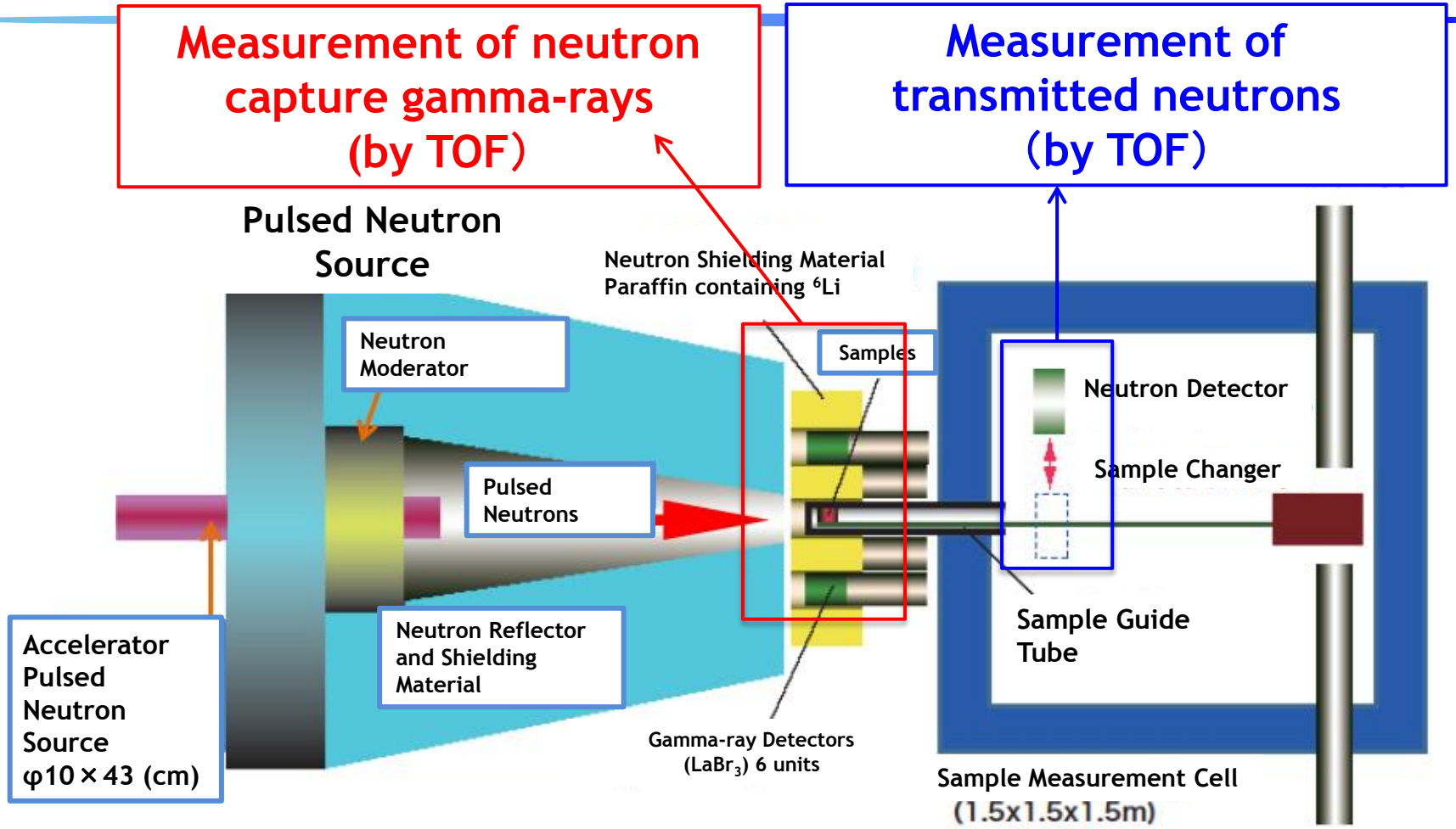

## Alternative to $^3$He neutron detection technology

**Expected outcome**
- Solving shortage of neutron detectors for nuclear security and safeguards

**Time of completion**
- During JFY2014



# NRD using NRTA and NRCA (3/3)




(Note) TOF = Time-of-Flight

The energies of the neutrons that interrogate the samples are given by the equation $E=(1/2)m(L/T)^2$, where L is the length of flight and T is the time of the detection of the neutrons or the neutron-capture gamma-rays (with the origin of time set at the time of the generation of the pulse of neutrons). The velocity (V) of the neutrons is given by V=L/T.

# Cooperation between EC/JRC and JAEA (2)

## List of specific topics to be studied in the area of cooperation described in the Agreement, May 2011

1. Nuclear safeguards and nonproliferation R&D

   In this area, the parties agreed to collaborate on:

   Safeguards by design
   Measurement using laser-based technology
   Process monitoring
   Environmental sample analysis

2. R&D in the area of illicit trafficking of radioactive and nuclear materials

   2-1 Detection
   2-1-1 - detection using neutron interrogation

   2-2 Nuclear Forensics
   2-2-1 - exchange of procedures
   2-2-2 - joint analysis
   2-2-3 - exchange of experts

3. Capacity-building, training, and technical support, in the areas of safeguards, security, and nonproliferation

   3-1 - curriculum development
   - exchange of lectures
   - co-sponsor training

   3-2 - coordination of technical support to third countries

Closing Plenary

J Tushingham, ESARDA Vice-President

Ladies and Gentlemen,

Welcome to the final session of the 35th ESARDA Symposium, in which I will attempt to summarise the key points from the week's meeting. I will try not to be too long, as I am sure that you would like to commence your journey home; or enjoy a final evening in this beautiful city.

For many of us, the week commenced with a series of meetings of the Working Groups, the details of which I hope will be available to those unable to attend via ESARDA's information repository of CIRCABC.

In addition to the technical working groups, the Editorial Committee discussed at length raising the visibility of the ESARDA Bulletin. We see this increasingly as a scientific journal, whilst the improved ESARDA website provides the opportunity for working groups to report on a range of activities on a shorter timescale. The Editorial Committee encourages authors to reference appropriate papers from within the ESARDA Bulletin, to promote the Bulletin to the wider scientific community as a free resource available via the ESARDA website.

The Steering Committee encourages organisations, active in research and development associated with the aims of ESARDA, to seek membership of the Association. During the meeting, the Steering Committee was pleased to approve an application for membership from Uppsala University: the University is particularly active in the area of non-destructive analysis and has worked with the IAEA through the Swedish Support Programme. At the same time, Michel Richard was accepted as an individual member following a distinguished career in safeguards and a number of roles within ESARDA.

The evening commenced with a meeting of the International Safeguards Division of INMM, during which Mr M Whitaker presented an excellent overview of US technology development under the Next Generation Safeguards Initiative (NGSI). The NGSI has brought about a high level of collaboration between US laboratories, capitalising on field trials with domestic and international partners, to bring about near-term testing and deployment. Ultimately, the initiative will provide practical support to the IAEA Department of Safeguards, including development of the State Level Concept.

Following an enjoyable reception, the first day of the 35th Symposium commenced with two plenary sessions. Our ESARDA President, Klaas van der Meer, opened the symposium, noting ESARDA's role in enhancing peace and identifying a number of threats and opportunities. 'On the bright side': disarmament verification and a broader interest in non-proliferation, with European Actions such as the Instrument for Stability. These contrast with threats from terrorist activity and the actions of States non-compliant with their safeguards

obligations. There were also neutral challenges, such as implementation of the Additional Protocol and the safeguards requirements on new facilities. An improved relationship between political and technical sciences was advocated.

We were honoured to receive an opening presentation from Mr W Bauwens, Special Envoy for Disarmament, on behalf of the Belgian Minister of Foreign Affairs. Belgium shares the vision of a world without nuclear weapons, but with effective security assurances. Disarmament and non-proliferation go hand-in-hand, with a call on India, Pakistan and Israel to join the NPT as Non-Nuclear Weapon States. However, we will not achieve universality of the NPT if we cause doubts ourselves.

On the subject of disarmament, Belgium welcomed and supported continued progress between the US and Russian Federation, whilst NATO membership and the newly-created nuclear non-proliferation and arms control committee enabled and promoted nuclear arms reduction.

Belgium strongly supports IAEA work in verification and implementation of an additional protocol, and welcomes improved efficiency and effectiveness of safeguards implementation. Mr Bauwens called on all Member States to meet the standards set by the EU, with a comprehensive safeguards agreement and additional protocol. He concluded by describing some of the safety and security challenges in the peaceful use of nuclear energy and urged Member States to uphold the NPT and not be distracted from it.

Mr N Whiting addressed the symposium on behalf of IAEA DDG Mr H Nackaerts. He described current challenges, including:

- how to continue to implement safeguards effectively; and the importance and limitation, in terms of its voluntary nature, of the Additional Protocol;
- how to deal with non-compliance, with the examples of DPRK, Syria and Iran; and
- rising demand for services, particularly with the growth in sensitive nuclear technologies.

Meeting these challenges required application of a holistic approach to the State as a whole – the State Level Concept – with increased safeguards-relevant information combined with State-specific factors, to be used in planning, implementation and evaluation, to seek credible assurance.

Cooperation between the Agency and State Authority was considered essential. The Agency will not discriminate – all States are equal with the law – but State cooperation will input into the implementation of safeguards.

Clandestine operations at undeclared facilities are seen as the main threat. More strategic targeting of resources is essential. Implementation must be focussed on areas of higher safeguards significance, changing working practices and deploying new technologies. The Department of Safeguards has published its first long-term plan, to invest in the best possible technologies. For the future, the Agency will persist with efforts for universal acceptance of

the Additional Protocol, with collaboration from States and increased focus on non-collaboration. More focused safeguards are foreseen, utilising all safeguards-relevant information, with robust measures for non-compliance.

The third presentation, by Mr Szymanski, EC ENER E, described current and mid-term developments within Euratom. He noted the challenge of additional requirements for resources, associated with new Member states, against a background of a reduction in resources of 5% over five years. An integrated management system was intended to improve efficiency, and to ensure internal compliance, whilst adopting ISO standards.

Euratom was reinforcing ties with Member States through annual bilateral meetings, meetings of all Member States and training events. Collaboration with the IAEA is intended to enable the drawing of independent conclusions whilst minimising duplication of effort. The EC and IAEA had agreed to establish a reflection group, to identify new areas for the Partnership Approach, following the production of facility-specific documents. The EC seeks swift implementation of improvements identified by the reflection group.

In his conclusions, Mr Szymanski noted that modern techniques and efficient approaches were required. Continued cooperation was essential, and ESARDA was seen as a unique vehicle for support.

The session concluded with Mr Fanghänel of the EC JRC ITU demonstrating the importance of nuclear security through the Seoul Security Summit; and the importance of safeguards and a reinforced role for the JRC through the 2013 symposium 'Benefits and limitations of nuclear fission in a low carbon economy'.

The Euratom 2014-2018 Programme includes at least three objectives related to safeguards. The JRC operates the On-Site Laboratories at La Hague and Sellafield, and also contributes through the European Commission Support Programme to IAEA Safeguards. A number of specific developments were described, including:

- liquid scintillator-based neutron measurements;
- sealing technologies;
- design information verification;
- strengthened safeguards (e.g. application of Large Geometry Secondary Ion Mass Spectrometry, LG-SIMS); and
- export control
- trade analysis and open source information.

Other activities included combating illicit trafficking through source attribution; support through the EU CBRN action plan; and education and training, through a European Nuclear Safety and Security School and events including an Additional Protocol training exercise for the IAEA.

The second plenary session welcomed the President of INMM, Mr Sorensen, who described recent developments within INMM, with a workshop on nuclear security scheduled for

October 2013 and participation encouraged in PATRAM 2013. A close ESARDA-INMM relationship is seen as critical to impact effectively on the issues associated with nuclear material management. Both parties are working to strengthen this relationship, with a letter of intent signed in 2011 and appointment of Wilhelm Janssens as the first non-US chair of the Chapter relations Standing Committee. There are currently 29 chapters, with strong international growth. In preparing for the future, INMM continues to engage with the IAEA, ESARDA and WINS, primarily through regional chapters. In conclusion, INMM was viewed as operating in a dynamic environment, filled with uncertainty and opportunity. A strong collaborative relationship between ESARDA and INMM is as important as ever.

Mr Meylemans, EC ENER, provided an overview of the state of play of Euratom Safeguards: adapting to new challenges. The Commission required transparency alongside efficient and effective use of resources. To this end, in 2011, the Directorate decided to implement an Integrated Management System, with clear objectives and risk management complemented by additional ISO standards for inspection, measurement and audit.

The evolution of cooperation with the Agency was explained, with Partnership Approach documents subject to update and incorporation of new technical developments into the Joint Use Agreement.

At the conceptual level, concern was raised that near-exclusive use of acquisition path analysis excludes some factors, such as a proven commitment of the State to non-proliferation.

The reflection group formed in 2011 had identified strategic areas for enhanced cooperation, and reported on how better to use the Euratom safeguards system. In summary, Euratom was looking for new opportunities to improve safeguards in the EU through partnerships and collaboration.

For the Agency, Mr Tsvetkov reviewed the history of cooperation with Euratom, and the improved efficiency under the new Partnership Approach of 1992. The 2011 reflection group identified underlying foundations and strategic areas of maximum impact, including more efficient management and use of common instruments, processes and procedures. Cooperation was strengthened with:

- introduction of the EC Quality Management System, with common elements to facilitate exchange of information;
- broader application of the principle ‘one job, one person’; and
- exchange and utilisation of safeguards information.

Enhanced cooperation in the longer term would include joint training and channels of communication, to ensure that cooperation leads to real gains in efficiency. Enhanced cooperation brought real benefits, with the reflection group identifying specific areas for greater cooperation and where more can be done.

Mr Senzaki, JAEA, described the background to establishment of the Integrated Support Centre for Nuclear Non-Proliferation and Nuclear Security (ISCN) in 2010. Support was provided in three areas:

- capacity-building assurance;
- assistance for infrastructure development; and
- technology development and support for detection and measurement of nuclear material.

Over two years, the first of these areas included:

- 16 courses or workshops on nuclear security;
- 9 courses on safeguards and State Systems of Accountancy and Control; and
- 8 courses on international non-proliferation.

A total of 679 international participants had attended these and four additional courses.

Support was also described to the research and development of technologies, with international collaboration from the US, IAEA and JRC. Overall, a range of international collaborations were in progress, with an ISCN workshop planned in conjunction with the IAEA conference on nuclear security.

Michel Richard brought the plenary session to a close with a review and vision for the future of ESARDA. The Association's historic objectives within research and development, and education, were successfully progressed, with working groups established, amalgamated or closed in response to changing requirements. A number of questions were put to the audience, associated with extending membership and collaborating with other think tanks.

...which brought nicely to a close a plenary session with a recurring theme for everyone to take forward of partnerships, cooperation and collaboration.

Of course, the major formal component of the ESARDA symposium is the technical sessions. It is impossible to do justice to the quality of the work presented by the authors within this closing plenary, but abstracts of the presentations and posters are available and papers will be published shortly on the ESARDA website.

Over 130 papers and posters were submitted to the Organising Committee, with over 60 organisations represented by the symposium participants. There were 21 technical sessions, with a dedicated poster session and, for the first time, a panel discussion: on disarmament verification.

There were 17 main themes, covering:

- Knowledge management and training;
- Safeguards by design;
- Implementation of safeguards;
- Fuel cycle back-end;

- Information management;
- Containment and surveillance;
- Integrated measurement and monitoring systems;
- Post-Fukushima: NDA techniques for molten fuel debris;
- Spent fuel verification;
- Neutron detection;
- NDA measurements;
- DA measurements; and
- Nuclear forensics.

Additional sessions dealt with synergies, partnerships and emerging technologies where ESARDA is increasingly working with other groups, covering:

- Non-proliferation, arms control & export control;
- Safeguards, safety and security;
- Disarmament verification; and
- Novel technologies.

A poster session gave an extended opportunity to present and discuss recent developments whilst, as already referenced, a panel discussion on disarmament verification highlighted technical challenges to the scientific community.

The Symposium benefitted from the support of a range of individuals and organisations, including:

- Staff of SCK-CEN, for organisation and administration;
- Staff of the Congrescentrum Oud St Jan, for their support and use of the Congrescentrum facilities;
- Commercial sponsors, for their financial support and provision of technical displays:
    - Nucleco
    - Urenco
    - Baltic Scientific Instruments
    - Canberra
    - Ortek
    - Ametek
    - Scannix
- Authors and speakers, for the high standard of papers and posters;
- Session chairs, for ensuring the smooth running of technical sessions;
- The Symposium Organising Committee; and
- All participants, for their attention and interactions throughout the symposium.

Papers presented at the 35$^{th}$ ESARDA Symposium will be made available to all participants via the ESARDA website, whilst a selection of papers will also be published in the ESARDA Bulletin. The Editorial Committee seeks volunteers to assist with the peer review of these papers.

Subject to confirmation, we expect the next annual meeting to continue the sequence of closed meetings in Luxembourg. Then, for 2015, we hope to secure a venue to follow on from the successful recent symposia, encouraging participants to leave their work in their hotel rooms and to meet informally to discuss and promote continuing developments in nuclear safeguards in an atmosphere of collaboration and cooperation.

And with that, I have the honour officially to close the 35$^{th}$ ESARDA Symposium. On behalf of the Symposium Organising Committee, I thank you all for your participation. Have a safe journey home, and goodbye for now.

# Modelling of high enriched uranium fission chamber with the code MCNPX

**A. Borella, R. Rossa, K. van der Meer**

SCK•CEN, Boeretang 200, B-2400 Mol
Email: aborella@sckcen.be

***Abstract:***

*Fission chambers loaded with high enriched uranium are used for spent fuel measurement in the so-called Fork detector. The Fork-detector is one of the work-horses during safeguards inspection and is also being considered for a verification of operator declared data in Go-noGo measurement station on spent fuel element in an encapsulation plant.*

*Having an accurate and validated model of the measurement equipment is beneficial for the investigation of this type of application. SCK•CEN is carrying out a significant effort to model the Fork detector with the MCNP code; however scarce information is known about the fission chambers.*
*This work describes the impact of the design information of the fission chamber on the calculated detector sensitivity and on the overall Fork response using Monte Carlo simulations.*
*The heavy ions transport in the active layer of the fission chamber was also studied and the resulting fission products energy spectra are compared with the available experimental data.*

**Keywords:** Spent Fuel; Fork detector; Non Destructive Assay, Heavy Ions transport; Fission chamber

## 1. Introduction

The Fork detector [1,2,3,4] is used by Safeguards inspectors for spent fuel measurements [5]. An accurate and reliable model of this detector would be beneficial in view of assessing and possibly reducing the uncertainties associated to spent fuel measurements.
An evaluation of the uncertainties is relevant for example in the case of the verification of operator declared data in a measurement station in an encapsulation plant [6] prior to disposal in a deep-geological repository.
SCK•CEN is engaged in the modelling of one type of such Fork detectors with the code MCNPX version 2.7.0 [7]. In a previous work [8], the developed model was validated with existing spent fuel data and it was stressed how important technical data associated to the fission chamber in use were missing.
In this paper, the fission chamber sensitivity is first defined and estimated with an approximated model. Then the same quantity is assessed by means of Monte Carlo calculations with the code MCNPX for several design variations. In addition the impact on the sensitivity of phenomena that are usually not accounted for, like the fission products transport in the active layer, are also considered and discussed. In the conclusion, the possibility to determine the neutron count rates by using a fully based Monte Carlo approach is summarized.

## 2. Simplified model calculations

The fission chamber sensitivity (or efficiency) is defined as the number of counts per second per unit of flux. It can be shown [9] that in ideal conditions the efficiency of the detector depends only on the amount of the fissile material. This however does not take into account the flux self-shielding and the self-absorption of the fission products in the fissile material layer. Under these assumptions, the efficiency can be expressed as:

$$\varepsilon = \sigma_{avg} \times 0.6022 \times \frac{m}{< A >}$$

Where σ is expressed in b, <A> is the atomic mass number and m is the mass in grams of the sample. The detector sensitivity depends via the average cross section on the flux shape. The knowledge of the flux shape is therefore very important when trying to benchmark simulations with experimental data.
The fission chamber in use in the Fork detector [10] considered here was the model FC167 from Centronic Ltd. For the two fission chambers used in the Fork detector [8,10] the sensitivity values are 0.122 and 0.119. These efficiency values were declared by the manufacturer and measured "in house" by comparison with the chamber 8819-51 which was calibrated using Mn foils in the Harwell GLEEP graphite reactor [3]. However, the measurement conditions and the distribution of the used neutron flux are not known. To determine the detector sensitivity, measurements in a well-known thermal neutron flux at the BR1 reactor [11] of SCK•CEN were carried out and a value of 0.136±0.006 was obtained [9].
In order to see if this value could be reproduced by Monte Carlo models and to understand what factors may affect the sensitivity, it was decided to build a model of the fission chamber only and to carry out a virtual experiment to assess the sensitivity that is obtained irradiating the detector with a neutron flux. The used design information of the fission chamber is presented in the next section.

## 3. Model and design information of the fission chamber

Despite the relevance of the information available in the technical data sheet, scarce information was available about the design of the fission chamber and the constructors were reluctant to disclose any additional details. As pointed out before, an important parameter affecting the efficiency is the mass of $^{235}U$ used in the fission chamber. The value of the $^{235}U$ mass (131.8 mg) used was taken from the Nuclear Material Accountancy records. In addition, based on the available information a model was developed accounting for:

- Stainless steel wall length: 203 mm
- Active length: 127 mm
- Diameter 25.4 mm
- Gas mixture at 4.5 atm, 98% Ar 2% $N_2$ [12]

As mentioned before, some technical design information was missing and more specifically:

- Material used in the coating of the active layer
- Thickness of the stainless steel wall and its composition
- Number of active layers and electrodes and their position
- Presence of other structural materials

A neutron radiography was carried out at the BR1 reactor of SCK•CEN [11], in order to gather more information on the design of the fission chamber. The radiography revealed that the fission chamber consists of two cylindrical parts one into the other. The thickness of the stainless steel walls of the cylinders is about 1 mm; there is a 1.5 mm spacing between the electrodes. In conformity with the specifications, it was supposed that both electrodes were made of stainless steel.

The number of coating layers could not be determined from the radiography. The fission chamber in use in our Fork detector resembles the design quoted in [13,14] where the cathode (outer cylinder) is coated and the anode (inner cylinder) not; however, it is also possible that two layers are used as indicated in [15, 16]. In support of this hypothesis, it was found [17] that the full model name of the fission chamber graved on the fission chamber is FC167/850/U235 and according to the coding in use [18], this would correspond to a coating area of 167 $cm^2$, a coating density of 850 $\mu g/cm^2$ and a 235U coating; this would correspond to 141.95 mg of coating material; if the enrichment of the coating material is 93%, the $^{235}U$ mass would then be 132.01 mg in very good agreement with the value taken from the Nuclear Material Accountancy. The value of 167 $cm^2$ is consistent with the presence of two active layers, one with a diameter of 23.4 mm (93 $cm^2$) and one with a diameter of 20.4 mm (81 $cm^2$).

It was decided to consider both the case where the uranium is coated on only one layer on the outer cylinder and the case where there are two coating layers. The main design variation was about the coating material; the following cases were considered, U metallic, $UO_2$ and $U_3O_8$. The latter seems to be the used form according to [13,15,19,20], while $UO_2$ is used according to [21, 22]. The

configuration with two coating layers was simulated only for $UO_2$. For this case it was assumed that the two coating layers had the same thickness.
Although the total mass of $^{235}U$ is the same, a difference in density and thickness may result in a different flux self-shielding and self-absorption of the fission products in the $^{235}U$ layer. To this extent, the number of U coating layers may play a role.
The corresponding considered characteristics are given in Table 2. In addition, design variations were considered in order to assess the impact of different design variations on the efficiency, in particular on the composition of the stainless steel.

| Coating Material | | $U_3O_8$ | $U_{met}$ | $UO_2$ | $UO_2$ (2 layers) |
|---|---|---|---|---|---|
| Density | $g/cm^3$ | 8.30 | 19.00 | 10.97 | 10.97 |
| Total coating mass | mg | 167.4 | 141.7 | 161.0 | 161.0 |
| Volume | $mm^3$ | 20.17 | 7.46 | 14.67 | 14.67 |
| layer radius | cm | 1.17 | 1.17 | 1.17 | 1.17 (layer1)<br>1.02 (layer2) |
| layer thickness | µm | 2.16 | 0.80 | 1.57 | 0.84 |
| layer area | $cm^2$ | 93.4 | 93.4 | 93.4 | 93.4 (layer1)<br>81.4 (layer2) |
| layer areal density | $mg/cm^2$ | 1.79 | 1.52 | 1.72 | 0.92 |

**Table 1:** Considered configurations for MCNPX simulations. The data for reference case are given in the first column on the left.

## 4. Detector sensitivity for the considered designs

A model of the detector in use was developed with the code MCNPX accounting for the geometry and the compositions of the materials used. The neutron flux was also modelled supposing a Maxwell-Boltzmann energy distribution with 293 K temperature and corresponding to the flux distribution of the BR1 reactor used in [9].
The sensitivity was obtained by using the F4 tally and the F4 tally multiplied by the fission cross section within the sensitive layer volume. The F4 tally represents the neutron fluence in given geometry cell per source particle. This approach assumes that every fission event results in a detectable fission signal; this is not always true due to phenomena such as the self-absorption of the fission product and a finite electronic threshold. The impact of such phenomena is discussed in the next section. A simulation without any material was also carried out and used to assess the incoming neutron flux.
From the simulations a value of the sensitivity of 0.201±0.001 cps/nv was obtained for the configuration with two $UO_2$ coating layers; the uncertainty is resulting from the statistical uncertainty associated to the MCNPX simulations. It was also found that the calculated sensitivity value does not change by changing the number of coating layers and the type of coating material.
The measured value of 0.136±0.006 is 32% lower than the calculated value. The calculated value is lower than the value of 0.223 cps/nv that is obtained from Eq. 1 using and average cross section of 661 b at 293 K. This difference can be explained with the presence of structural materials that attenuate the incoming neutron flux. For example, it was found that structural materials, such as the stainless steel wall, and their thickness affect the sensitivity (the used one mm thickness reduces the sensitivity by about 5%). Simulations with different composition of stainless steel were carried out and the resulting sensitivity changed by less than 4%. The lowest value was obtained with a 316 stainless steel with Manganese amount of 2% and the highest with a FeC steel without any other materials.

## 5. Other phenomena and their impact on detector sensitivity

In the results presented so far, it was assumed that every fission event results in a detectable fission signal and that the simulated fission rate relative to the incoming neutron flux was representative of the detector sensitivity.
When determining the fission rate in a cell with MCNPX, the geometry and used materials are accounted for and the full neutron transport is carried out. Therefore, effects as self-shielding are taken into account in the simulations. However, not all fission events result in a detectable pulse due phenomena that are not accounted for in the simulations such as [13]:

- self-absorption of fission products in the active layer,
- wall effect, due to fission products emitted by the layer but not depositing all the energy in the gas (e.g. due to the cylindrical geometry),
- electronic threshold, rejecting the fission products that do not deposit sufficient amount of energy,
- incomplete charge collection, electrode spacing and fill gas effect.

Therefore, one expects that the simulated fission rate with the method presented in Section 4, overestimates the actual sensitivity due to the presence of the abovementioned phenomena. In the next sections, the impact on the fission chamber sensitivity of these variables is discussed.

## 6. Self-absorption of fission products in the active layer

Due to the finite thickness of the U layer, not all the fission products (FP) escape the layer and deposit sufficient energy to induce a detectable signal. To estimate the number of FP that escape the layer and their distribution in energy, an approach based on MCNPX was used, using its capability to carry out heavy ions transport [23]. The heavy ions transport in MCNPX is based on stopping powers determined by the Bethe-Bloch formula [24] with Sternheimer and Peierls density effect correction [25]. All ions are assumed to be fully ionized and there is no implementation of partial charge states or charge pickup in the current version of the code. The Vavilov model [26] was used for charged particle straggling.

The attenuation of the neutron flux due to $^{235}U$ for neutrons with 25.3 meV for the configurations considered in Table 1 was determined to be 0.26%. However, in case of a fission in the resolved resonance range, it is expected that a significant self-shielding is present; by Monte Carlo methods, it was calculated that, during a measurement with spent fuel [8], the share of fission with energy above 1 eV is 6.0±0.6%. It was therefore assumed that the self-shielding can be neglected and uniform source of fission products within the active layer was simulated. Further, an isotropic emission of the FP was assumed. In the simulation, the atomic (Z) and mass (A) number of FP were sampled first based on the fission yield (FY) taken from the ENDF/B-VII.0 data library [27].

The FP kinetic energy is then sampled from an energy distribution which is a function of the sampled mass number A. This distribution has been obtained from total kinetic energy as a function of the mass number, the so-called TKE(A) distribution [28], applying momentum conservation and assuming that the FP are emitted at 180 degrees.

It was found that FP with a cumulative fission yield corresponding to 99.7% of the cumulative fission yield obtained with the data in [27] could be transported in MCNPX. This is because only a set 2200 heavy ions can be transported; some fission products (e.g. $^{92}Kr$, $^{150}La$) are not included in this list; on this basis, it was decided to remove those heavy ions from the distribution used in the simulations.

An intrinsic limitation of this approach is that it was not possible to transport both fission fragments within the same MCNPX event; instead each fission fragment was transported independently. The bias introduced by this approximation is discussed in [9] and is an overestimation of the sensitivity by 1.3 % for the one layer and 2.1% for the two layer $UO_2$ configurations, respectively.

The cases shown in Table 1 were considered. The results of the MCNPX simulation indicate a loss of 0.7% of fission fragments due to the fission fragments leaking through the basis of the cylindrical shell used to represent the active layer; this result is a consequence of the FP sampling, which is uniform in a cylindrical volume, and the number of FP emerging from the inner surface is slightly lower compared to the number of FP emerging from the outer surface.

It was also found that the loss due to self-absorption of the fission fragments is 5.2% in the case of one layer in $U_{met}$, 19.6% in $U_3O_8$, and 16.6 % in $UO_2$. When two coating layers of $UO_2$ are used the self-absorption is 6.9%.

The obtained energy distributions of the fission fragments emerging from the layer is shown in Figure 2.

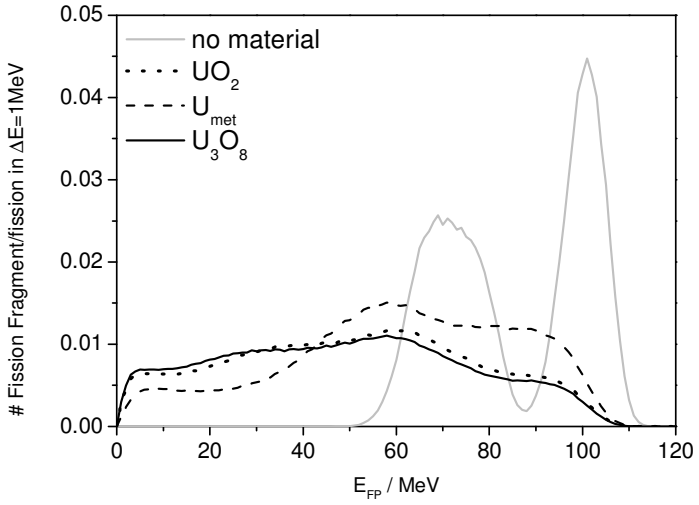



**Figure 1:** Number of fission fragments per fission in a 1 MeV energy bin as a function of the fission fragment energy. The plot shows the fission fragments emerging from the layer surface and entering the detector volume for no material and one layer of $UO_2$, $U_{met}$ and $U_3O_8$.

## 7. Energy threshold, wall effect, electrode spacing and filling gas

To have a detectable event, it is not sufficient that a FP emerges from the coating layer with a minimum kinetic energy. It is also necessary that an energy above a threshold is deposited in the sensitive volume of the fission chamber.

To reduce the background due to noise, gamma rays and alpha particles a non-zero threshold is applied when carrying out measurements with the fission chamber. In particular, a threshold of 27 mV was chosen in the preamplifier [22]. With a capacity of 2.2 pF, this corresponds to $5.94 \times 10^{-14}$ C of charge. Assuming 28 eV as mean ionization energy for Argon [13], this corresponds to an energy of 10.4 MeV.

The energy deposition of fission fragments in the filling gas was also calculated with MCNPX (F8 tally). In this way it is possible also to account for fission product that do not deposit their entire energy within the active volume (e.g. they hit again the electrode or U layer).

The developed model of the fission chamber should account for all of the effects related to the particle transport. However, it should be mentioned that phenomena affecting the formation of the electric pulse (e.g. incomplete charge collection) and those depending on the design of the fission chamber, cannot be modelled. These effects are difficult to estimate and cannot be determined with the data and publications available. However, according to the test carried out the detector operates in the plateau region and therefore in optimal condition to minimize these effects.

If we look at the particles depositing at least 10 MeV in the gas, the results of the MCNPX calculations reveal that the loss of fission events amounts to 9.3% in the case of one layer in $U_{met}$, 26.1% in $U_3O_8$, and 22.6 % in $UO_2$. When two coating layers of $UO_2$ are used the loss is 11.3%.

The resulting energy deposition spectra in the active layer, in the gas and the internal stainless steel for the case with one coating layer of $UO_2$ were determined. The obtained results indicate that the FP lose most of their energy in the $UO_2$ layer; however most of them deposit energy also in the filling gas. It was found that the percentage of the FP that was depositing at least 1 MeV was 99.9 % in the $UO_2$ layer, 83.3% in the gas, and 40.7% in the SS anode.

The calculated and measured energy deposition spectra are also compared in Figure 3. The experimental data were arbitrarily normalized both in amplitude and response to match the peak in the simulated data. Although an absolute comparison is yet to be done, Figure 3 reveals a similar shape between the spectra.

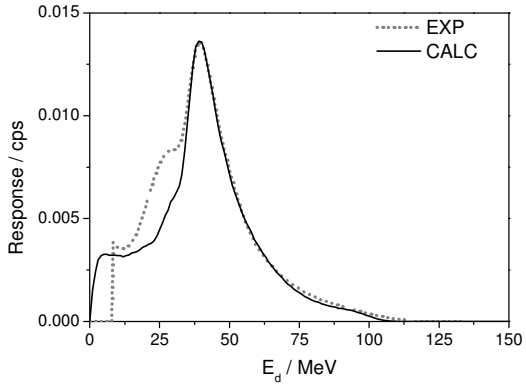



**Figure 2:** Measured and calculated energy deposition spectra in one layer of $UO_2$.

The obtained results about the loss of fission fragments are lower when compared with the data given in [13] that reports, for a 10 MeV threshold, a loss of 29% for a coating thickness of 1.0 mg/cm$^2$ and of 41% for a coating thickness of 2.0 mg/cm$^2$ in a parallel plate counter. Such an underestimation can be due to the fact that in MCNPX the heavy ions are assumed to be fully stripped and there is no implementation of partial charge states or charge pickup at this time. The impact of such a phenomenon can be the subject of a further investigation.

## 8. Conclusions

An MCNPX model of the fission chamber in use in the Fork detector [1-4,8] was built. This model was used to determine the efficiency of the chamber in presence of a flux distributed according to a Maxwell-Boltzmann distribution. The impact of several design parameters on the sensitivity was assessed.
In addition, ad-hoc calculations were carried out to verify the transport of fission products in the coating layer and fill gas and their impact on the fission detection efficiency. Several configurations were studied and the overall impact on the efficiency determined. Depending on the coating layer thickness this effect can be from about 10 to about 25 %, slightly below the calculations given in [13].
The results presented in this paper are in line with the results at thermal energies in [9], where the bare detector efficiency was determined from measurements in a well characterized neutron flux; in that case, the measured sensitivity in [9] is about 32% lower than the one obtained from Monte Carlo simulations and the self-absorption and detection of FPs can account at least partly for this deviation.
The obtained energy deposition spectra also were compared with available experimental data, although there was no attempt for a comparison on an absolute energy deposition scale.

## 9 Legal matters

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# Mobile Information Technologies and Managed Access During On-site Inspections

Karl Horak, Sandia National Laboratories
Mailstop 1371, PO Box 5800
Albuquerque, NM 87185
E-mail: kehorak@sandia.gov

## Abstract:

Managed access has practical application in a variety of arms control and nonproliferation situations. Typically this has involved shrouding, partial access, constrained viewing, random sampling, and so forth. However, since the inception of managed access in the late 20thcentury, information technology has developed powerful new tools that can be combined with on-site inspection to good effect. In this paper, the author will explore the use of mobile communications and computing systems (smart phones, tablet computers, iPads, etc.) in managing access during on-site inspection activities. The list of mobile technologies that could be applied include augmented reality, real-time access to databases, situationally-aware information feeds, inspection team tracking, GIS apps, enhanced communication channels (texting, microblogging, social media), and novel sensor packages. Interesting use-cases exist for escorts, site owners, hosts, inspectors, remote participants, and OSI trainers. The list of hurdles to implementing mobile as a tool for managing access include: the device approval processes, limiting functionality based upon site sensitivities, accurate geopositioning inside structures, controlling data flow (both inward and outward), dealing with cellphone dead zones/lost signals, synchronizing with off-site databases, battery life, and jamming/spoofing/hacking issues. Because most mobile devices are capable of communicating via cellphone channels, Wi-Fi, and Bluetooth, they are potentially intrusive on all three levels. Managed access scenarios will be explored to estimate the impact of mobile technologies on a safeguards complementary access inspection, a CTBT inspection, a putative FMCT inspection, and a CWC-like challenge inspection.

**Keywords:** on-site inspection, mobile computing, information technology

## 1. Introduction

Mobile computing technology is making tremendous strides, whether smartphone (iPhone, Droid, Blackberry) or tablet computer (iPad or Android). These devices are inexpensive, small in size and weight, wearable, and rapidly become ubiquitous. Usage of mobile for Internet access has tripled in the last two years[1]. With powerful sensor capabilities now standard on current models, there are many opportunities for field exploration.

However, there is a gap in the application of current mobile technology to arms control and safeguards, especially in on-site inspection (OSI). Obviously, there are situations where safety and security make the use of mobile devices impractical, but this author contends that many of these hurdles can be overcome.

Critical to the successful use of mobile devices in OSI is its ability to add value to the inspection process while addressing safety, security, and operational concerns. A key element of OSI is managed access and in this paper the author will examine how mobile technologies can fit into that paradigm. Bear in mind that both inspectors and hosts can profit from the effective use of mobile computers, in the field and during training.

## 2. Managed Access Techniques

OSI regimes since the 1980s to verify major arms control and safeguards agreements have established unprecedented level of cooperation and openness, added stability and confidence, and provided new verification tools to deal with proliferation priorities today[2]. One OSI concept that has been central to its success has been managed access, the ability of the inspected state party to protect sensitive

information not related to the treaty or agreement while demonstrating compliance.

A brief listing of commonly employed managed access techniques is given in Table 1 [3].

| | |
|---|---|
| Shrouding | Partial or complete covering of an object or area. May be conformal or non-conformal |
| Protecting data | Logging off computer systems; removal of sensitive papers |
| Denial of access | Prohibiting entry or viewing |
| Partial access | Access permitted with limitations, such as allowing visual inspection of the interior of an enclosed space from its entrance |
| Restricted sampling | Analysis only for the presence or absence of relevant materials |
| Random access | Permitting access to only a random percentage of available areas (laboratories, bunkers, etc.) |
| Alternative information | Providing other collaborative evidence of compliance without granting access |

**Table 1. Managed access techniques**

These OSI techniques clearly predate the use of cellphones or other wireless devices. None-the-less, implementing a mobile component can speed up the inspection process, enhance transparency without compromising security, and offer significant improvements over current radio communications.

## 3. Mobile Technologies

Smartphones and tablet computers bring a wide variety of information management tools to the OSI party. Both host and inspector may profit by their use, but the host has more flexibility in that they can use site-specific pre-approved hardware systems that meet their facility's safety and security constraints. Similarly, the host is likely to create dedicated apps and webpages just for mobile host team members.

On the other hand, inspection teams will have generalized requirements specific to the treaty regime in question and their mobile information systems can offer targeted solutions. However, there may be significant difficulties to overcome before an inspected state party will permit external mobile devices into a sensitive site.

The more obvious technologies for use during OSI with managed access are summarized in Table 2 [4].

| | |
|---|---|
| Inspection team tracking | Using secure types of FourSquare-like software |
| Augmented reality | Superimposing information and images upon the camera field of view |
| Enhanced communication channels | Text messaging, microblogging, social media, image sharing |
| Real-time database access | Information available on demand/as needed |
| GIS applications | Mapping, orientation, familliarization |
| Novel sensor packages | Either intrinsic COTS sensors or add-ons |
| Situationally-aware information feeds | Data transmission based on location, activity, and context |

**Table 2. Applicable mobile technologies**

Inspection team tracking could be performed automatically or with a FourSquare-like check-in procedure. At significant waypoints, the user (host or inspector) would check-in and have their GPS coordinates, time-date stamp, photographs, and comments logged onto a central system. Team tracking could be unilateral (host only or inspection team only) or bilateral (location information shared to all participants).

Augmented reality (AR) is the process of superimposing additional data upon a computerized view of a real-world scene. Commonly this is done with visual elements added to the view as seen with a smartphone's camera. Whether simple graphical elements, supplementary text, or complex multimedia presentations, the processing is done in real-time and adds to or augments one's perception of a scene[5].

One can easily imagine this sort of system providing an escort with visual safety warnings or an inspector with overlaid imagery from previous visits. Recently AR has received considerable publicity with the announcement by Google of their wearable heads-up AR display, Google Glass[6].

Mobile devices can provide enhanced communications due to their multi-channel capabilities—Wi-Fi (WLAN), Bluetooth, and cellular radios. While two-way radios form the backbone of current escort and inspection team

communications, the ability of mobile computers to provide more than voice transmission is an important strength of theirs. SMS services for text messaging and microblogging (Twitter-like), social media services for sharing observations and events (Facebook-like), and image sharing services (Flickr-like) could be possible with a properly configured and secured cellphone or tablet.

Where wireless service is reliable enough, realtime database access could be provided. This might be facility-supplied local information sent to the escort team, for example, chemical hazards and safety data. The information could be from an IAEA database (either public or restricted) shared with the inspection team or it could be from some other source and supplied to all parties during an inspection. Many use cases can be envisioned.

GIS applications allow mobile users to accurately know their location, elevation, and orientation. IAEA and UN inspectors in Iraq in the early 1990s were sometimes misled by their minders. Stand-alone GPS units were needed to confirm inspection locations [7]. Similar tools in cellphones can track one's inspection route for later playback or analysis. Again, either host, inspector or both could make use of this technology[8].

Cellphone technology continues to draw innovation from a wide number of developers, both professional and hobbyists. Novel sensor packages appear frequently, for example, a simple radiation detector based on a cellphone camera with its lens covered[9] or a spectrometer made by adding a diffraction grating over the lens[10].

Another exciting possibility is situationally-aware information – data that is sent to one's phone or tablet based on sensed activities. Mobile devices can "know" whether their user is moving or stationary, keying in information or browsing webpages, using the camera at a particular location facing a particular direction, and many other states, even the light level the user is experiencing.

## 4. Hurdles to Implementation

Needless-to-say, there are numerous hurdles to be overcome before mobile devices are allowed into the onsite inspector's toolkit. Considering that treaty regimes usually tightly specify the equipment that inspectors will be permitted to use, device approval will be a critical step. Of course, site owners may easily authorize their escort's use of mobile, but the inspection team will not usually have this freedom. Already though, the boundary between cellphones and other common devices is blurring, for example, a Bluetooth-enabled hearing aid.

Mobile devices, to be effective as we have outlined in a previous section, have to enable a number of features. Battery life becomes an important consideration when continuously operating energy hogs like screen illumination, GPS, background processes, and camera.

Currently GPS can be useless inside structures, especially multi-story metal structures. (Even cellphone reception can be blocked in interior spaces.) Advances are being made to develop chips that use one's last known GPS position plus direction, gait, and time to estimate location within a structure.

Lost signals/dead zones are a fact of life in normal cellphone use. Many facilities of interest to onsite inspectors are remote and may have intermittent communication service at best. Apps need to be designed to fail gracefully, collect information when offline, resume uploading when back online, and operate with some onboard capability even when out of network.

Controlling data flow to mobile computers is another aspect in need of consideration. Data volumes could be very high, especially if streaming video or realtime sensor data is involved. Apps will need to be built to throttle their data appetites, whether sending or receiving, so that critical realtime functions are not slowed or halted.

Similarly, synchronization with off-site databases has to be performed such that needed information is available in a timely manner while avoiding overloading available bandwidth. Downloads to offsite servers could be postponed until after the inspection, uploads could be performed ahead of time, and only transactional records posted in realtime.

Intrusiveness is probably the largest hurdle to overcome. Allowing these small but powerful computers into a facility will require forethought and planning, even when they are under host control.

Information security will always be a concern with any onsite inspection technology. Information authentication, validation, verification, and other aspects of security

cannot be overlooked in these situations. Sensitive information must only be accessible to authorized recipients, data streams must be protected from accidental or purposeful alterations. The possibility of communication disruption could result in a failed inspection.

In many use cases a solution to the intrusiveness of a mobile device is to use “crippled” devices, for example, one with its camera and Wi-Fi disabled. While this solves the intrusiveness problem, it removes useful capabilities. Instead of permanently crippling the camera, can we develop a photo app that disables video recording when within a defined area? Can we trust that app to work flawlessly even if the GPS signal is lost? In any case, inspected state parties will find more-or-less COTS mobile devices useful for their host teams.

Finally, even if we grant that mobile computers are allowed under certain circumstances, policies and procedures have to match the maturity of the technology. The use of social media during an inspection, for example, posts to Facebook and Twitter, will have to be conducted under appropriate guidance and with suitable user training.

## 5. Scenarios

Given the above discussion of managed access, mobile technologies, and implementation hurdles, it may be worthwhile to reflect briefly on specific use cases.

The use of mobile technologies during a complementary access inspection under an Additional Protocol would require IAEA Board approval of such and consultations with the State concerned. That said, the IAEA has the right to make use of internationally established communications systems, including satellite systems and other forms of telecommunication. This opens the door to the use of current mobile devices in many ways.

Challenge inspections include inspections of non-declared facilities[11]. Inspections of non-declared facilities are known by different names in different treaty documents. In the chemical weapons agreements they are specifically called challenge inspections[12], while in START, they are termed 'special right of access visits." During a 2009 CWC challenge inspection exercise at Kirtland Airforce Base, cellphones and Wi-Fi devices were used extensively, but they were not permitted in limited areas of Sandia National Laboratories.

FMCT inspections are still theoretical pending an actual treaty. It's entirely conceivable that in the course of negotiations and the writing of implementation legislation that modalities will be included that allow for the use of state-of-the-art mobile technologies.

Similarly, actual CTBT inspections await entry into force, but probably represent one of the more plausible scenarios. Due to the potentially enormous inspection perimeter (enclosing an inspectable area of 2000 sq. km.), cellphone communications may be important, not only in accomplishing the inspection mandate but for the safety of the host and inspection teams involved. Considering the remoteness of any putative test site, wireless coverage may be extremely limited or non-existant. Novel technologies such as portable, rapidly deployed cell towers could play a role.

## 6. Conclusions

Managed access is an accepted part of many international agreements and treaties. Although the specifics vary widely among different regimes, the general concept of protecting information unrelated to verification while demonstrating compliance is commonly agreed to.

Smartphones and tablet computers bring a wide variety of novel capability to onsite inspection activities. Both host and inspector may benefit from their use. The host has more control because they can use site-specific pre-approved systems, which meet their facility’s safety and security constraints. The host will be able to create dedicated apps and websites strictly for host team members who may be using cellphones and tablets.

Inspection teams will have generalized requirements tied to the treaty regime in question and their mobile information systems can offer useful IT solutions. However, there may be significant difficulties to overcome before an inspected state party will permit external mobile devices into a sensitive site. Perhaps host-provided systems, crippled or not, can allow international inspectors to make use of mobile technology without compromising host site security.

As mobile devices become smaller, more common, and more frequently embedded in other objects, it will become important that OSI modalities consider the impact of cellphones and their ilk on managing access during inspections. The next generation of arms control and safeguards inspectors, being digital natives, will insist on having these useful tools as part of their kit.

Not to be forgotten is the usefulness of these technologies during OSI training. Adaptive, mobile, perhaps wearable systems will allow trainees to experience virtual or augmented facilities and interact within them. This can provide both inspectors and hosts with an unprecedented level of realism in preparing for compliance verification during onsite inspections.

## 7. Acknowledgements

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# WHAT DO WE MEAN WITH KNOWLEDGE MANAGEMENT?

**T. Jonter (1), W. Janssen (2), M. Marin Ferrer (3)**

(1) T. Jonter, Chair of the WG Training and Knowledge Management, Professor of International Relations, Stockholm University, Stockholm, Sweden. E-mail: Thomas.jonter@ekohist.su.se

(2) M. Marin Ferrer, Vice-Chair of the ESARDA WG Training and Knowledge Management, Scientific Officer, Nuclear Security Unit at the Joint Research Centre European Commission Institute for Transuranium Elements, Ispra, Italy

(3) W. Janssen, Institute for the Transuranium Elements Joint Research Centre, European Commission, Via Fermi, Ispra 21020 (VA) Italy, E-mail: willem.janssens@ec.europa.eu

**Abstract:**
*The main purpose of the work conducted by the working group Training and Knowledge Management (TKM) is to improve education and training in safeguards and non-proliferation for students and professionals in the European Union. The scope of the task was very clear when the TKM was established in 2004: to deal with training and education in safeguards and non-proliferation. However, what was and still is not clear is in what way this task is connected to "Knowledge Management". The term "Knowledge Management" has never really been problematized and in the beginning of 2011 during a TKM meeting it was decided that the term and its meaning should be a central issue to be addressed at an ESARDA symposium. The purpose of this paper is twofold. Firstly, the intention is to give a general background of the term knowledge management, its roots and emergence as an academic discipline and how it is used by practitioners in the field. Secondly, based on how knowledge management has been used in academia as a theoretical discipline and in the work life as a practical tool-box, the intention is to discuss how TKM ought to deal with the term. In other words, how could knowledge management be an integrated and natural ingredient of TKM´s activities and thereby contribute to the overall goal of ESARDA to enhance the management of safeguards within Europe.*

**Key words:** Knowledge management, training, education, safeguards, non-proliferation

## 1. INTRODUCTION

The main purpose of the work conducted by the working group Training and Knowledge Management (TKM) is to improve education and training in safeguards and non-proliferation for students and professionals in the European Union. The TKM was created by ESARDA in 2004 with the aim to set up an annually course in safeguards and to develop course syllabus to reduce the education deficit in the safeguards area. The scope of the task was very clear and well defined from the beginning. However, what was and still is not clear is why a part of the WG´s name is "Knowledge Management". The term "Knowledge Management" has never really been problematized and during a TKM meeting in the beginning of 2011 it was decided that the term and its meaning should be a central issue to be addressed at an ESARDA symposium. The authors of this paper were given the task as chairs and members of TKM to prepare a presentation of the topic in order to enable an informed discussion and to define what TKM means with the term.

The purpose of this paper is twofold. Firstly, the intention is to give a general background of the term knowledge management, its roots and emergence as an academic discipline and how it is used by practitioners in the field. Secondly, based on how knowledge management has been used in academia as

a theoretical discipline and in the work life as a practical tool-box, the intention is to discuss how TKM ought to deal with the term. In other words, how could knowledge management be an integrated and natural ingredient of TKM´s activities and thereby contribute to the overall goal of ESARDA to enhance the management of safeguards within Europe.

## 2. THE EMERGENCE OF THE DISCIPLINE KNOWLEDGE MANAGEMENT

The concept of knowledge management has its roots in management thinking and practices. In a broad sense, the subject deals with organizations and their various resources and how they they can perform better. In most cases, definitions of the term emphasize the importance of organisational behaviour and especially how certain strategies and practices can be used by private and public organisations to identify, create, and distribute various recourses in order to enable adoptions of insights and experiences. Key elements in this understanding of the concept are the focus on the knowledge process and its capability to spread the gained knowledge and experience to different layers within the organization [1].
The above understanding of the concept – there are plenty more in the vast literature on knowledge management - beg for an obvious question: is it really dealing with something qualitatively new or is only a new variation of an old theme? The answer is both yes and no. It can be argued that the issue how to make organizations to perform better is an old and classic theme in the management literature. A central question in the academic and non-academic management literature is how different approaches and methods can be used to improve the performance of staff and organizations. On the other hand, it is also possible to defend the standpoint that the knowledge management as it has been defined over the last twenty years is a qualitatively new phenomenon. In 1991, knowledge management was introduced as an academic discipline by the Japanese scholar Hirotaka Takeuchi at the Hitotsubashi University. During the same year courses in knowledge management were taught at New York University. The Swedish company Skandia was the first corporate organization in the world to have an employee with the title “Chief of Knowledge Management” when Leif Edvinsson was hired in the beginning of 1990s. Whatever school of thought or definition preferred, KM has a common ground or consists of a couple of key aspects that are viewed in an holistic approach: people, processes, technology (or) culture, structure, are all necessary parts that constitutes an organization wholeness [2].Another interesting school of thought is analyzing and dividing knowledge in mainly two dimensions, tacit knowledge and explicit knowledge. In this framework tacit knowledge is identical to internalized knowledge that each individual or staff member is not aware of, for example how certain work tasks are carried out. Explicit knowledge, on the other hand, is form of knowledge that is used in an aware and often learned way that can be communicated and explained to other individuals inside and outside the organization. According to conducted research, successful organizations are successful because they have been able to transform internalized tacit knowledge into explicit knowledge and vice versa if needed with the result that the organization is performing better. Through different processes where the organization´s various resources are aligned in such a way that an interrelated dynamic is created. There are of course other KM models where other specific interrelated main forms of knowledge are stressed [3].

With a risk of oversimplification, KM encompasses a set of theoretical approaches and best practices models whit the ambition which can be summarized in a couple of central points.

What is an organization and how can an organization align its resources in order to achieve certain goals?

- How can an organization perform better? – maximize the (intangible and tangible) assets of the organization
- Knowledge is central: share information, combine tacit and explicit knowledge, combine knowledge in a system (organization) and individual capabilities to learn, transfer of new knowledge (innovation).
- Strategy and technology (or culture) are important tools in KM processes.

After this brief and perhaps too simplifying overview, it is easy to draw the conclusion that TKM has not been dealing with knowledge management.

## 3. HOW CAN TKM DEAL WITH KNOWLEDGE MANAGEMENT?

Generally speaking, the authors of this paper see three different answers to the question how TKM should deal with knowledge management. The first solution represents perhaps the easiest way out: we shouldn´t deal with KM at all since it is rather a task for each individual company or public organization to decide how they could perform better since they know best what they possess in terms of financial, technological, staff resources. TKM has neither the competence nor the organizational and financial resources to play an active role in such ambitious efforts. In other words, the solution is then to take away "Knowledge Management" from the name of the WG and just call it "Working Group for Training and Education".

The second solution would be to admit that the KM is too important to ignore for ESARDA given its central goal to assist and help nuclear organizations within EU to improve the capability and capacity to manage safeguards and non-proliferation tasks. But how, is then, of course, the vital question. What should the ambition be for ESARDA? One solution, if this way of arguing is to be followed, would then be to establish a new WG with the goal to deal specifically with this issue; organize conferences and workshops, analyze the needs for KM in different organizations etc.

A third approach – which we personally prefer – would be to deal with KM within TKM but with limited and well defined tasks. In this context, we will give two examples of tasks which could be characterized as typical KM activities and simultaneously would not be too ambitious for TKM to deal with.

### 3.1 Maintaining historical knowledge/Information gathering.

In this task the aim is not only to carry out courses in safeguards and non-proliferation but also to systematically produce and update course material – text books, course compendium and digital versions. Another task within this scope could be to disseminate course material to regulatory authorities and universities which are planning or running courses in safeguards. Even if it could be argued that TKM is already doing this since a text book has been published (Ed. Greet Janssens-Maenhout, *Nuclear safeguards and Non-Proliferation*, European Communities, 2009). However, the point here is that this goal can be enhanced to include a more systematic assessment of the present existing literature on safeguards and non-proliferation in terms of what can be used in training and education. In addition, the TKM can present these assessments on the NuSaSET´s web site to reach out to universities, companies and organizations (http://www.nusaset.org/ ). In this context, it can be worth mentioning that TKM has during this symposium discussed the possibility to historical research on the emergence of the European Commission´s safeguards systems which could be used in future courses in safeguards and non-proliferation.

### 3.2 Collaborative efforts

Another TKM oriented activity could be to create and deliver courses in collaboration between ESARDA, EU regular authorities and universities. A good example is the 8th ESARDA course in Nuclear Safeguards and Non-Proliferation held at Uppsala University 12-16 September 2011. The course was conducted as acollaboration between ESARDA, the Swedish Radiation Safety Authority (SSM) and Uppsala University. TKM could present the one week course that has been designed and developed since 2005 in cooperation with SSM that joined the project with staff members and experts. The SSM staff members set up a team which contributed with everything from financial means to teaching. Uppsala University contributed with teachers and conference room. This course was by and large identical to the ESARDA course in Ispra, but the idea here is to take a step further and design *specialized training courses* for certain targeted groups that could be of interest. For example, one course could be focused on *Diplomats,* especially EU diplomats, who want to learn more about the connection between peaceful nuclear energy

and nuclear weapons programs and especially how the safeguards mechanisms are working in practice and about other tools within the NPT regime as export control and physical protection.

Another targeted group could be to give specialized courses to *States* with limited or no nuclear experience which have plans to build nuclear power plants or start nuclear energy programs. TKM could in cooperation with other organization tailor-fit a course that would suit the state per se in their early ambitions to learn more about safeguards and non-proliferation. In fact, Joint Research Center (JRC) organised a duplication of the ESARDA training and education course on nuclear safeguards and non-proliferation in Kuala Lumpur, Malaysia, on 25 February – 1 March 2013. This event, requested and agreed upon with Atomic Energy Licensing Board (AELB) already back in 2011, was organized in the framework of the AA219636 with DG DEVCO. This compact course was open to masters degree students, in particular nuclear engineering students, but also to young professionals and International Relations/ law students. The course aims at complementing nuclear engineering program and courses by including nuclear safeguards in the academic curriculum. This course was dedicated to participants from ASEAN countries, in order to increase knowledge and understanding in the non-proliferation and safeguards field and to build capacities in countries that are envisaging moving to nuclear power in a near future. The course gathered 37 participants from Australia, Philippines, Myanmar, Thailand, Malaysia and Vietnam. The lecturers came from South Korea (KINAC), US Texas University, Japan Atomic Energy Agency (JAEA), IAEA, Australian ANSTO and the University of Liege.

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[2] Nonaka and H. Takeuchi, *The Knowledge-Creating Company: How Japanese Companies Created the Dynamics of Innovation* (Oxford, Oxford University Press, 1995)

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# Change in impurities observed during the refining and conversion processes

**Peter Button, George Healey**

Canadian Nuclear Safety Commission, Ottawa, Canada
E-mail: peter.button@cnsc-ccsn.gc.ca
george.healey@sympatico.ca

***Abstract:***

*In 2009 the Canadian Safeguards Support Program undertook a sampling campaign to follow a batch of UOC feed material through the refining and conversion processes at Cameco Corporation's Blind River and Port Hope facilities. A total of 59 samples were taken in three categories as follows; sixteen UOC samples, twenty-two samples during the conversion of UOC to UO3, and twenty-one samples during the conversion of UO3 to UF6. Samples were analyzed for trace element content by ICP-MS, uranium and lead isotope ratios by MC-ICP-MS and hydrogen and oxygen isotope ratios by IRMS techniques.*

*The analytical results have been examined to find the extent to which trace element levels and isotopic compositions change at the various processing steps. Both national forensics organisations and IAEA safeguards are interested in characterising these materials.*

*Initial results were presented at the ESARDA 2011 meeting in Budapest, but the data was incomplete and the results from different laboratories were inconsistent. In addition, the apparent increase in some trace elements during some stages of conversion was thought to be erroneous. This paper will provide an update on our findings.*

**Keywords: UOC; uranium; impurities; forensics; traceability; isotopic**

## 1. Introduction

The principal objective of this work is to be able to identify the origins of Uranium Ore Concentrates (UOCs) and the down-stream uranium products leading to $UF_6$. Characterisation of these materials is of interest to the IAEA in validating declarations. Establishing the origins of these materials is of great interest to national forensic programs and has application to IAEA safeguards. This program was conducted as a joint IAEA task; samples were shared with the IAEA (Seibersdorf Analytical Laboratories, SAL) and JRC-ITU.

For a variety of reasons, the distribution of samples and the delivery of analytical results has taken took far longer than anticipated. Initial results from the laboratories were incomplete and often inconsistent. We now have more complete data sets at our disposal.

First we will review the results for UOCs that were discussed in the earlier paper. Secondly, an analysis of the changes that occur during refining and conversion is provided.

## 2. Collection of samples

Canada hosts uranium mines, a refinery and conversion facility. Collectively, they provide the possibility of tracking the same material from ore through to $UF_6$; though in this case, we only collected samples from Uranium Ore Concentrate (UOC) through to $UF_6$.

Samples were collected at Cameco's Blind River refinery in November 2009 and at Cameco's Port Hope conversion facility in February 2010. The location of these two facilities is shown in Fig. 1.

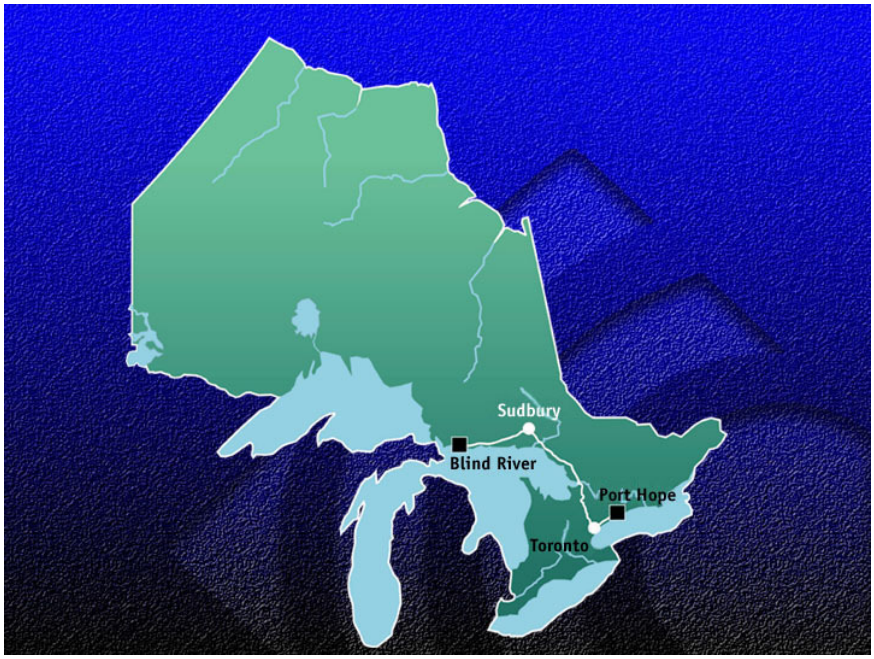


**Figure 1 - Refinery and Conversion facilities in Canada.**

Figure 2 indicates the UOCs comprising the blend and the proportion of each UOC entering the refinery feed. Blending is necessary to ensure a consistent product. As we will see later, the impurities present in each UOC vary considerably; blending helps to even out these variations, but it also makes it difficult to track a subsequent uranium product back to a particular UOC.

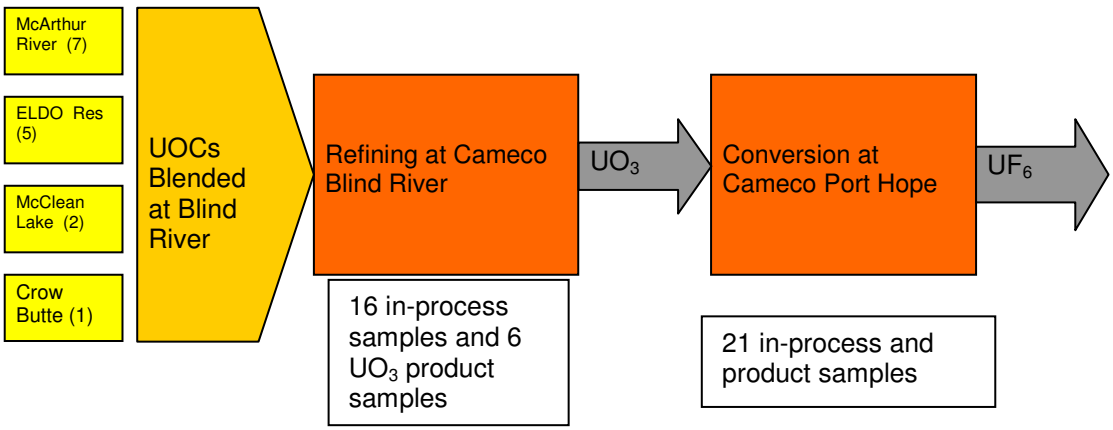


**Figure 2 Major processes: blending, refining and conversion**

Considerable effort was expended during the sampling campaigns to ensure that samples would be taken from the same material as it progressed through the refining and conversion processes. Samples were carefully homogenised before splitting to ensure that the samples distributed to the various laboratories would be consistent.

Samples were distributed to:

- Queen's University Facility for Isotope Research (QFIR), Kingston, Canada
- Seibersdorf Analytical Laboratory (SAL), Vienna, Austria
- Institute for Transuranium Elements (ITU), Karlsruhe, Germany

## 3. Elemental impurities in UOCs

The Blind River refinery receives UOC from several sources with different milling practices. The sources of UOC involved with this sampling campaign were:

- McArthur River mine – processed at Key Lake Mill – UOC as U3O8
- Eldo Resources – processed at Rabbit Lake Mill – UOC as UO4●2H2O
- McClean Lake mine – processed at McClean Lake Mill – UOC as U3O8
- Crow Butte – ISL – processed at Crow Butte Central plant – UOC as UO4●2H2O

Impurities in the UOCs result from the source material, mining practices and the milling processes. The results reported in 2011 are mostly still valid, but here we offer some explanation for the Trace Elements (TE) behavior.

### 3.1 McArthur River UOC

McArthur River is a high-grade underground uranium mine. The average ore grade is 15-16% $U_3O_8$, the principal uranium minerals being pitchblende and coffinite. Due to the high U content, special procedures are necessary for radiological health and safety.

Ore is mined by the "raise bore" method. The raise bore technique requires tunnels (levels) to be driven above and below the ore body. A pilot hole is drilled between levels. When the pilot drill breaks through into the lower level, the drill bit is removed and a larger cutting bit, a reamer head, is attached to the drill string. As the reamer head is retracted toward the upper level, the cuttings drop to the lower level floor and are subsequently transported to an underground grinding station. The cuttings are wet ground to create a slurry which is then pumped to surface.

For mining purposes, reamed out holes (raises) are backfilled with Portland cement-based concrete. To extract as much ore as possible, the raises overlap, thus some of the backfill concrete placed in one hole is removed as the next hole is drilled. Material shipped to Key Lake consists of ore, concrete, and waste rock in slurry form.

At Key Lake, the slurry is blended (down blended) to a concentration of 4% $U_3O_8$. Low-grade U ore and other mineralized waste from previous open pit mining operations is prepared (ground) for use as down blend material. After downblending, leaching, solvent extraction, precipitation and calcining. the final product is $U_3O_8$. Note that it is already a blended product due to the down blending. Other impurities may be found due to the introduction of Portland cement in the mining process.

The consequence of the addition of so much dilution of the mined ore is that the trace element spectrum of McArthur River/Key Lake origin UOC will be more representative of the dilution material than of the ore. The use of $H_2SO_4$ and sulphate chemicals will result in S contamination in UOC.

For samples analyzed, the average total TE content of McArthur River UOC is 10935.6 ppm. The total average S level is 8872.8 ppm and the total Mo is 1231.3 ppm. These two elements constitute 92.4% of the total TE content. Subtracting S and Mo from the total leaves 831.5 ppm for all of the other trace elements. Thus, except for the high values for S and Mo, McArthur River UOC is high quality (high purity) material. Elements which unambiguously distinguish the McArthur River UOC from the other

three UOCs (marker elements) would be S, W, B, and Ca. However, the S values were the result of temporary production problems and are subject to process changes. The Ca values could also be influenced by blending changes. The W and B values derive from mining practice and would be more difficult to change. Thus, W and B are considered to be the more reliable markers for McArthur River UOC with respect to this set of four UOC sources.

## 4.2 Eldo Resources (Rabbit Lake) UOC

Formerly an open pit mining operation, ore is currently extracted from the underground Eagle Point deposit. Eagle Point is an underground hard rock mine where low-grade ore (0.6-0.7 % U3O8) is mined by conventional vertical blast hole stoping methods. Because the ore is low-grade it can be mined and processed by conventional methods thus, only normal dilution of the ore takes place. Mined ore is transported to the surface for processing at the Rabbit Lake mill. Processing at the mill consists of grinding, $H_2SO_4$ leaching, decantation, solvent extraction, gypsum precipitation, and precipitation of UOC as uranium peroxide (UO4●2H2O). Wear of the grinding balls will add Fe. Use of alloy steel balls could contribute Si, V, Cr, and Mn.

The average total TE content is 23832.5 ppm (just over 2%). 34 elements are present at a level of 1 ppm or greater. Seven elements are present at levels greater than 1000 ppm: Al, Ca, Fe, Mg, Mo, S, and Si. Of these, Ca, Mg, and S are probably present due mainly to process chemistry. Al, Fe, Mo, and Si would be ore-related as carryover through solvent extraction; similarly for the high levels of P and K. The very high Al value probably reflects the presence of Illite, a complex alumino silicate clay mineral, in the ore zones.

13 elements could be considered to be markers for Rabbit Lake UOC, Al, Cr, Cu, Fe, Ga, Mg, Mn, P, Pb, Sc, Si, Ti and V. There is a notable presence of rare earth elements exemplified by 6 ppm Y and 3.5 ppm Sc.

## 4.3 McClean Lake UOC

Mining at McClean Lake ceased in 2010 and the McClean Lake mill (JEB mill) shut down in June 2010. At McClean Lake, ore deposits were found near the surface and were exploited by open pit mining methods. These deposits are located about 8 km northeast of the Rabbit Lake mine. Ore supply to the mill in 2009 and until shutdown was mainly from the Sue E deposit together with a small amount from the Sue A and Sue B deposits. The average grade mill feed in 2009 was 0.97% U3O8. Upon receipt at the mill, ore is finely ground, slurried with water, and pumped to the leaching circuit. In the leaching process, sulphuric acid and hydrogen peroxide are added to the slurry to dissolve the uranium from the ore. After solvent extraction purification, uranium is precipitated as ADU and calcined to U3O8.

The average total TE content is 5318.5 ppm, just over 0.5%. 26 elements were present at a level of 1 ppm or greater. Two elements are present at levels greater than 1000 ppm, Na and S with Mo present at an average level of 975.9 ppm. The presence of the high levels of Na and S are due mainly to process chemistry. Rare earth values at McClean Lake are more comparable to those at McArthur River than at Rabbit Lake.

Ni, Th, Zn, and Zr – are potential markers for McClean Lake UOC. All four elements originate from ore, none are process-related.

## 4.4 Crow Butte UOC

Crow Butte is a solution mining or in-situ recovery (ISR) operation located in Nebraska, USA. At Crow Butte, a sodium bicarbonate leach solution with O2 injection passes through the porous ore zone where mineral dissolution takes place and is then pumped to the surface via production wells.

Product solution passes through resin filled ion exchange columns. Uranium and trace elements transfer from the product solution to the resin. Loaded resin is stripped with an aqueous solution of sodium bicarbonate ($NaHCO_3$) and salt (NaCl). Uranium is present in the strip solution as uranyl

carbonate, $UO_2(CO_3)_3$. Following further chemical processing, uranyl peroxide, ($UO_4 \bullet 2H_2O$) is precipitated, thickened, dewatered, washed, and dried and drummed.

Coffinite is the main uranium mineral species. The origin of the uranium is in-situ rhyolitic ash material. The most common minerals in the sandstone are quartz, plagioclase (calcium alumino silicate), K-feldspar, coffinite, pyrite, calcite, illite/smectite and tyuyamunite [Ca(UO2)2(VO4)2•nH2O]. Garnet, magnetite, marcasite, and illmenite ($FeTiO_3$) are present. Vanadium mineralization is present.

The average total TE content is just under 1.5%. 25 elements are present at a level of 1 ppm or greater. Five elements are present at levels greater than 1000 ppm: Na, V, Ca, Zr and S. The high Na level is a result of process chemistry. All other trace elements originate from the ore.

The high level of V is a result of the presence in the ore body of tyuyamunite and possibly unspecified species in volcanic ash. The high level of Ca could have numerous mineral sources principal among which would be calcite. The high Zr value probably originates from garnet, particularly taking into account the notable Hf value of 11.3 ppm. The high S value would originate from the pyrite content of the ore. The total for the five trace elements in excess of 1000 ppm is 12864.4 ppm (90.95% of the TE total). There is notable presence of yttrium (7.9 ppm) which, considering the level of P (233.3 ppm), might indicate the presence of xenotime (YPO4) in the ore.

Four elements – Hf, Na, V, and Zr – are potential markers for Crow Butte UOC. Sodium is process-related; Hf, V, and Zr originate from ore.

## 5. Refining and conversion processes

The steps required to convert UOCs into $UF_6$ and the corresponding availability of TE measurements are as follows:

- Receipt of UOCs - TEs measured
- Blending of UOCs - TEs not measured
- Digestion - TEs calculated and used a reference level
- Purification (solvent extraction) - TEs measured
- Concentration (boildown) ($HNO_3$ > UNH) - TEs measured
- Denitration (UNH > $UO_3$) – TEs measured
- Shipment of product $UO_3$ and receipt at Port Hope – TEs measured
- $UO_3$ preparation – TEs measured
- $UO_2$ production – TEs measured
- $UF_4$ hydrate production – TEs measured
- $UF_4$ dehydration - TEs measured
- $UF_6$ production -TEs measured

The first conversion activity at Blind River is to prepare a feed solution for the solvent extraction purification process. Receipts of UOC (lots) from different sources have different TE contents. It is desirable to supply a feed of reasonably uniform composition to solvent. In addition to UOC, steel shot and other scrap metal together with $UO_3$ dust recovered from the denitration dust collectors is used as a blend component. Once the drums in a blend are dumped into the feed hoppers and transferred to the digester the UOCs essentially lose their identity. Any product characteristics are derived from the blend and subsequent processes.

Digestion converts solid UOC, plus additions, into a liquid feed solution for solvent extraction. UOC is continuously fed into the digester where it is converted to uranyl nitrate. UOC is not the only material entering the digestion system. For operational purposes iron is added either as grit blast or as a solution from scrap recovery. Process chemicals such as phosphoric acid may also be added as well as re-cycled sumpage.

Following digestion the solution is fed to the solvent extraction circuit to produce purified uranyl nitrate referred to as “OK Liquor”. The purified uranyl nitrate (OK Liquor) is concentrated to produce uranyl nitrate hexahydrate (UNH). The concentrated UNH is then fed to the denitration pots for thermal decomposition to $UO_3$, the final product of Blind River.

At the Port Hope facility, pulverised $UO_3$ is fed into fluid bed reactors where $UO_3$ is reduced to $UO_2$. $UF_4$ production is a two-stage operation. The first stage is reaction of $UO_2$ with HF in aqueous solution to produce uranium tetrafluoride hydrate ($UF_4 \bullet 2.5H_2O$). Dried and pulverised $UF_4$ is then fed to the $UF_6$ flame reactors to produce the final product.

## 6. Trace Element Changes During refining and conversion.

The following charts (Fig 3 to Fig 14) demonstrate the changes in specific TEs during the refining and conversion processes. The reference level (100%) is the value calculated based on the blending information and the reported values for the individual UOCs. Note the scale on left (green) covers the full 100% and the scale on the right (blue) is set to accommodate the much lower values encountered later in processing. As expected, the effect of solvent extraction is to substantially reduce TE values in the digester input feed. Reduction factors are different for different elements and range from 1 to 1700.

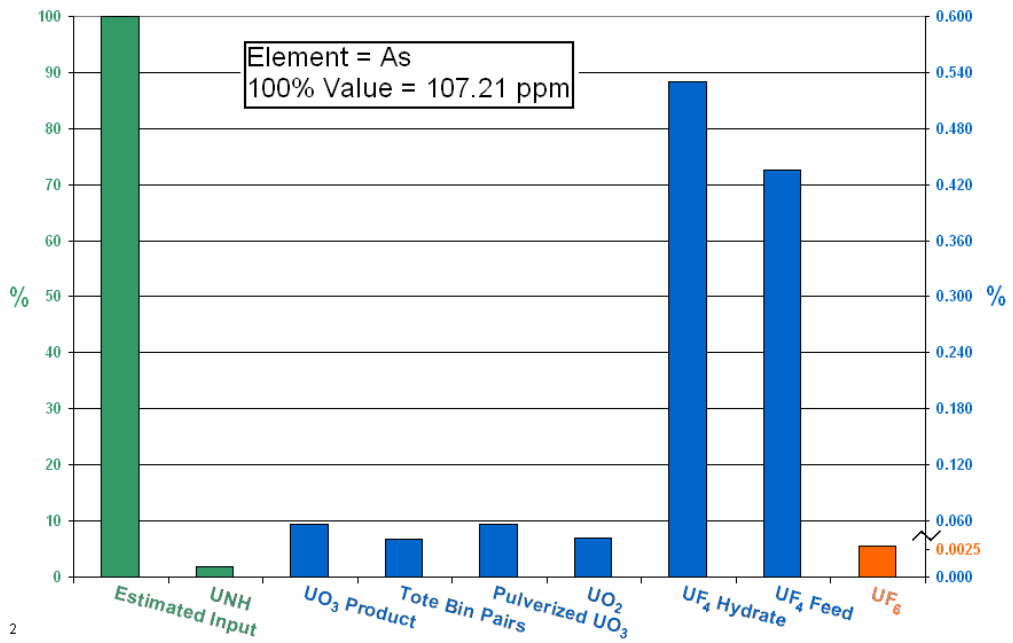


**Fig 3. Arsenic returns in $UF_4$ production, but removed in final product (below detection limit)**

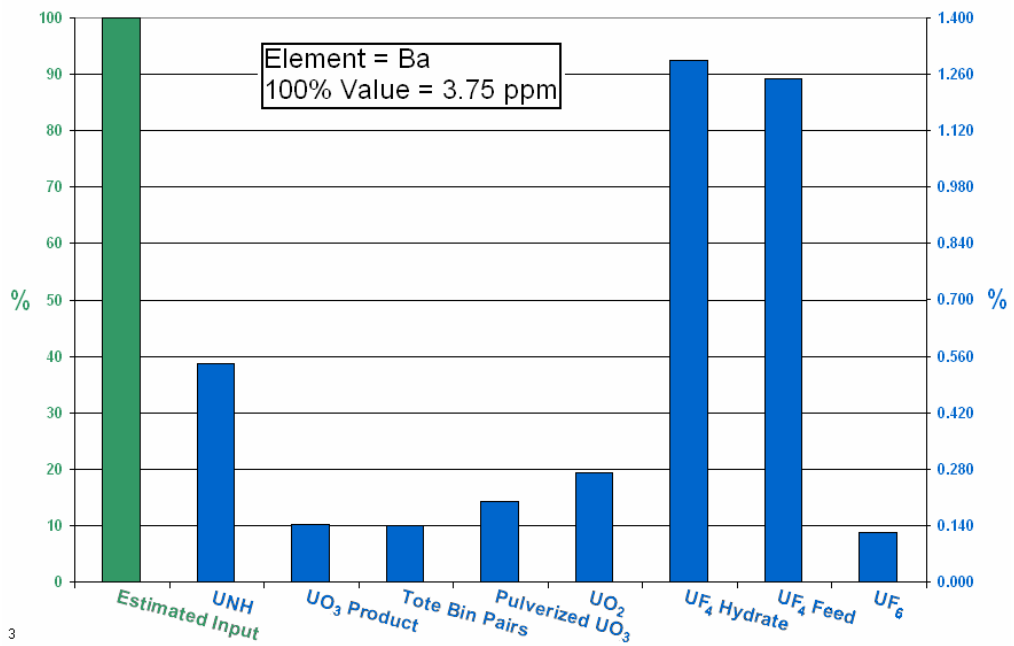


**Fig 4. Ba increases during UF4 production, but reduced in final product.**

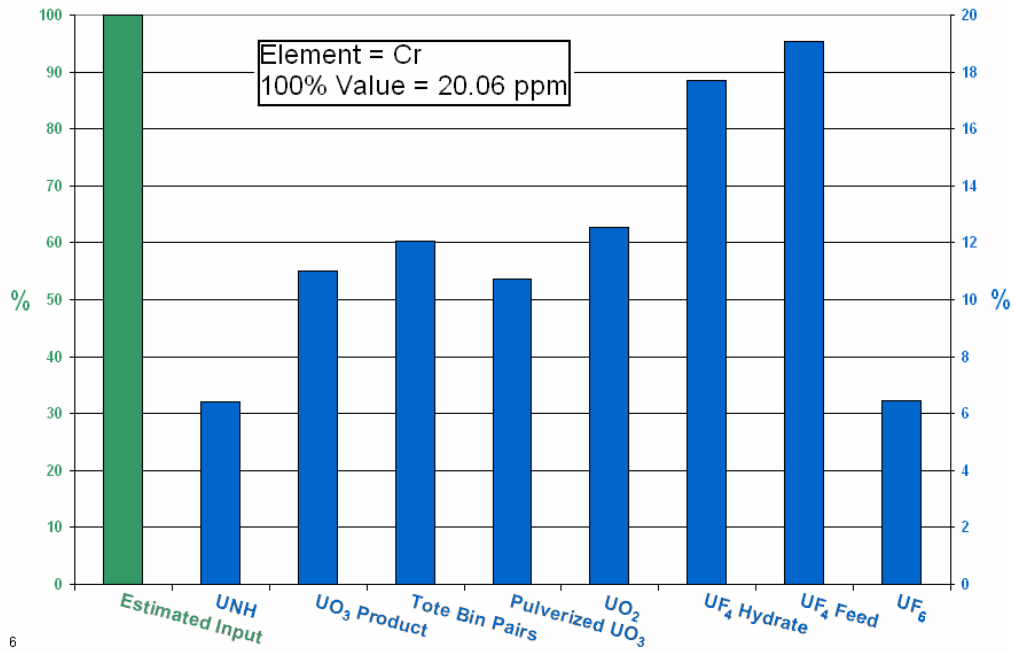


**Fig 5. Cr increases after UNH – evidence of corrosion**

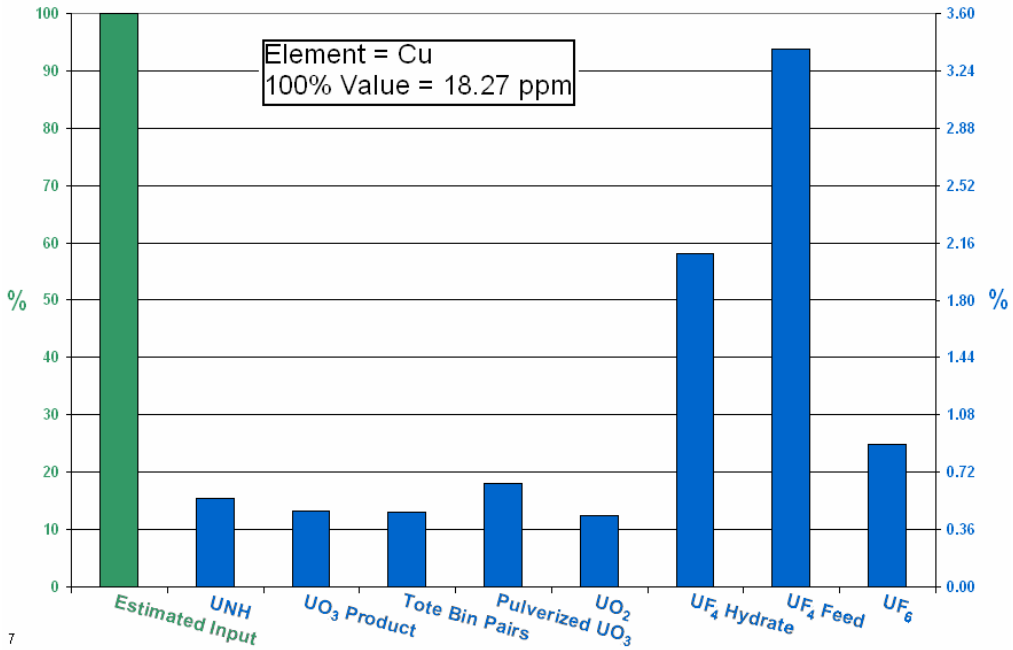


**Fig 6. Cu increases in UF4 production**

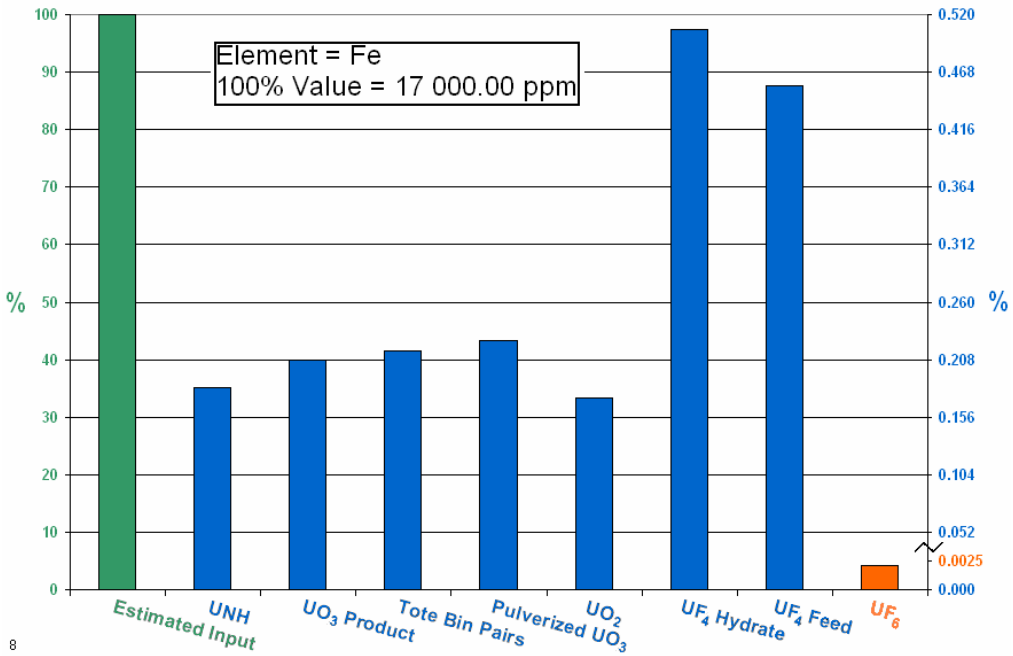


**Fig. 7 Evidence of corrosion in $UF_4$ production. Note iron may be added in digestion.**

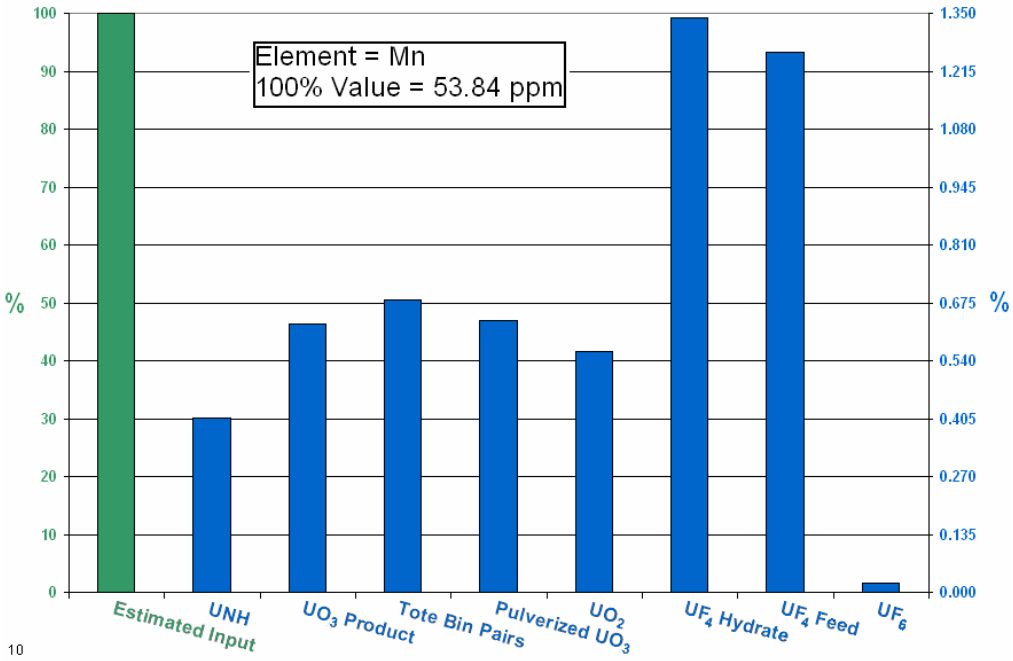


**Fig 8. Evidence of corrosion during $UF_4$ production**

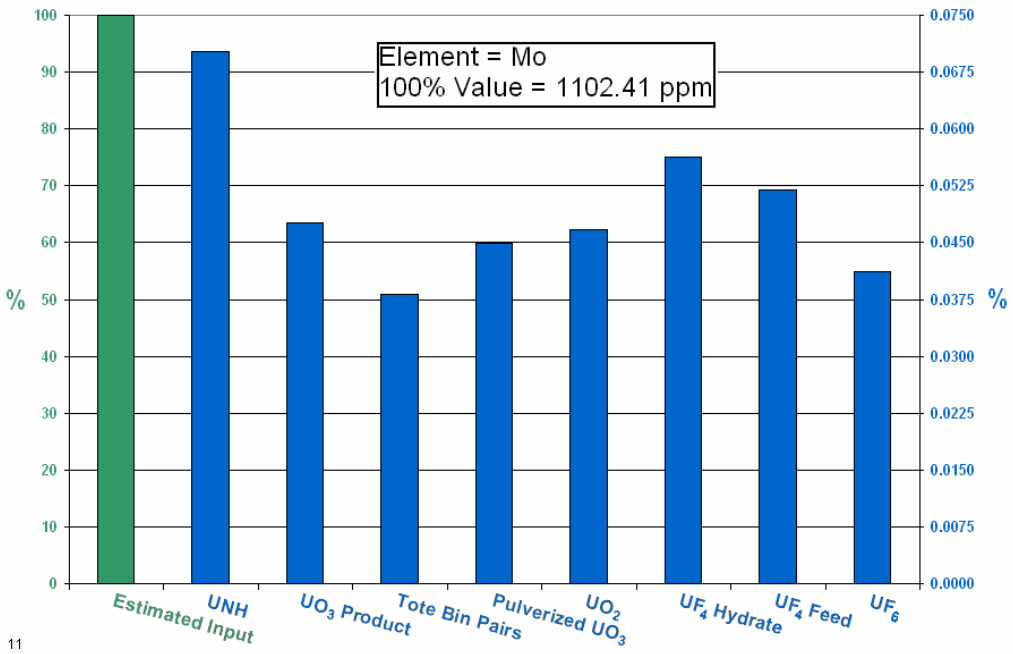


**Fig 9. Mo is difficult to remove completely**

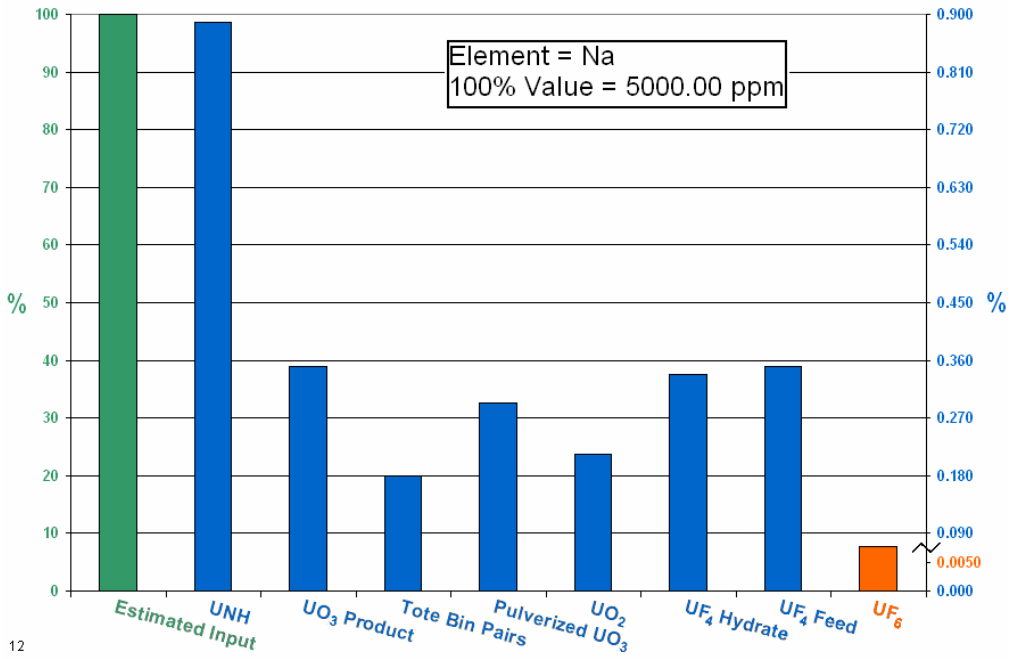


**Fig 10. Na removed to below the detection limit**

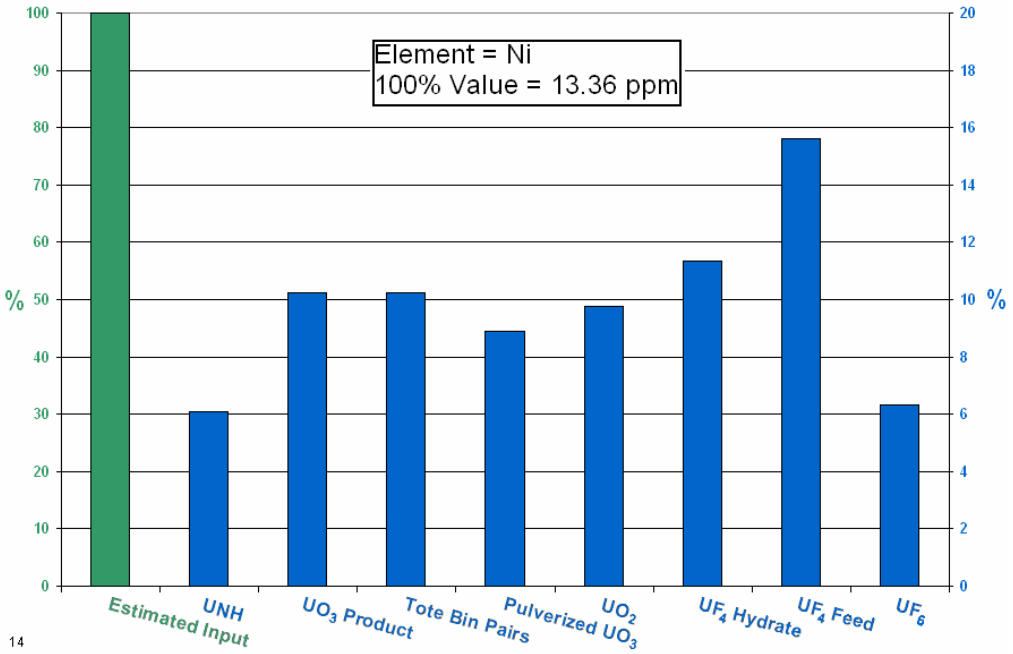


**Fig 11. Difficult to remove Ni completely**

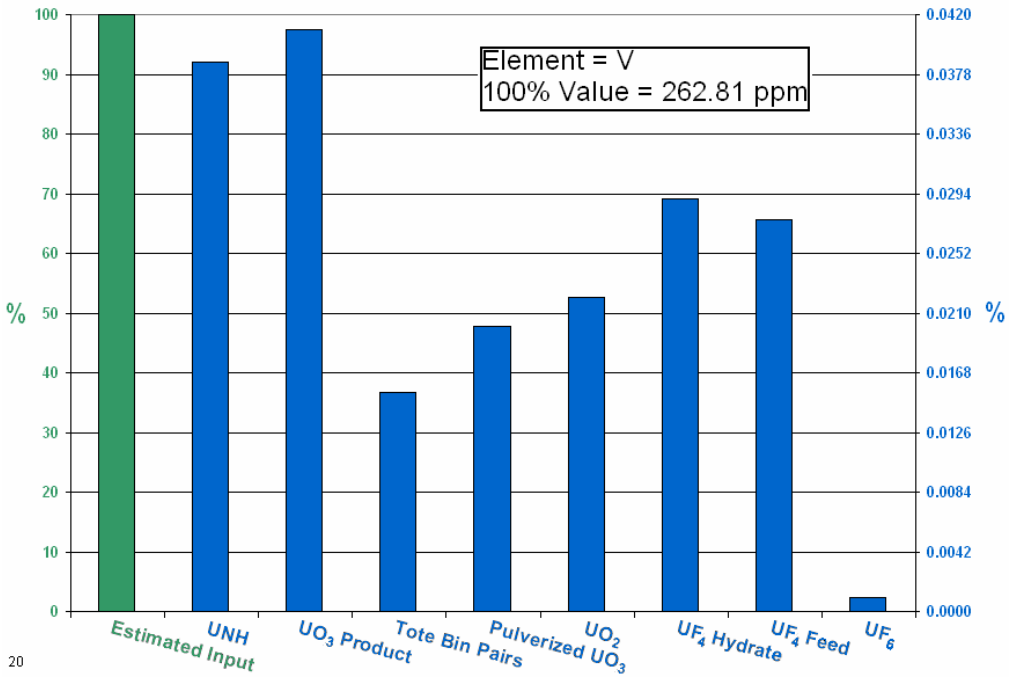


**Fig 12. V is common in ores**

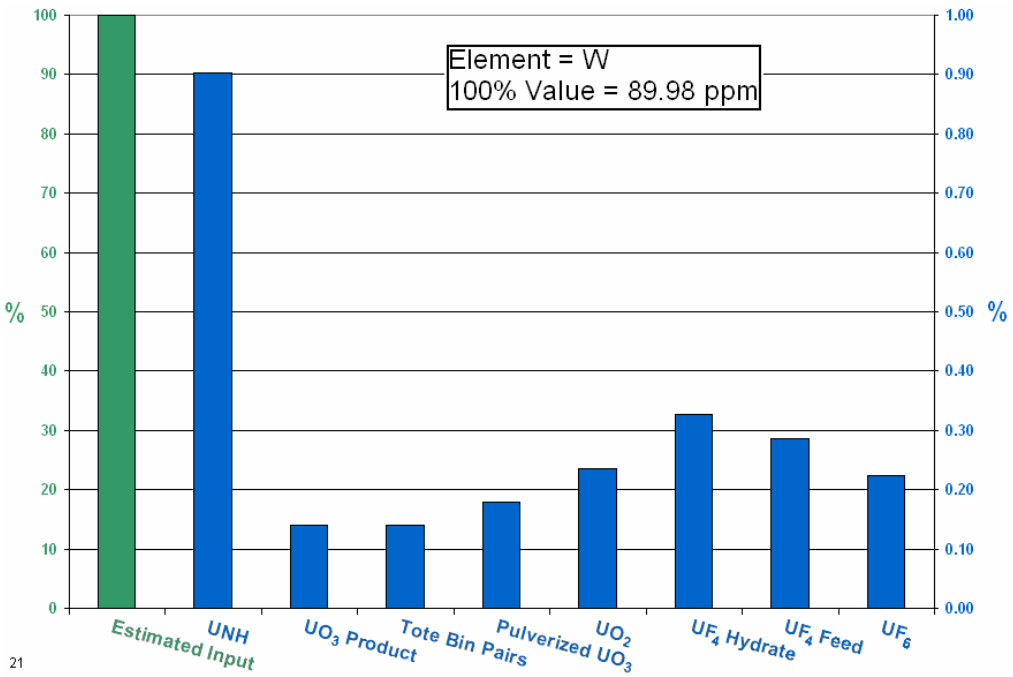


**Fig 13. Tungsten is often found in ores**

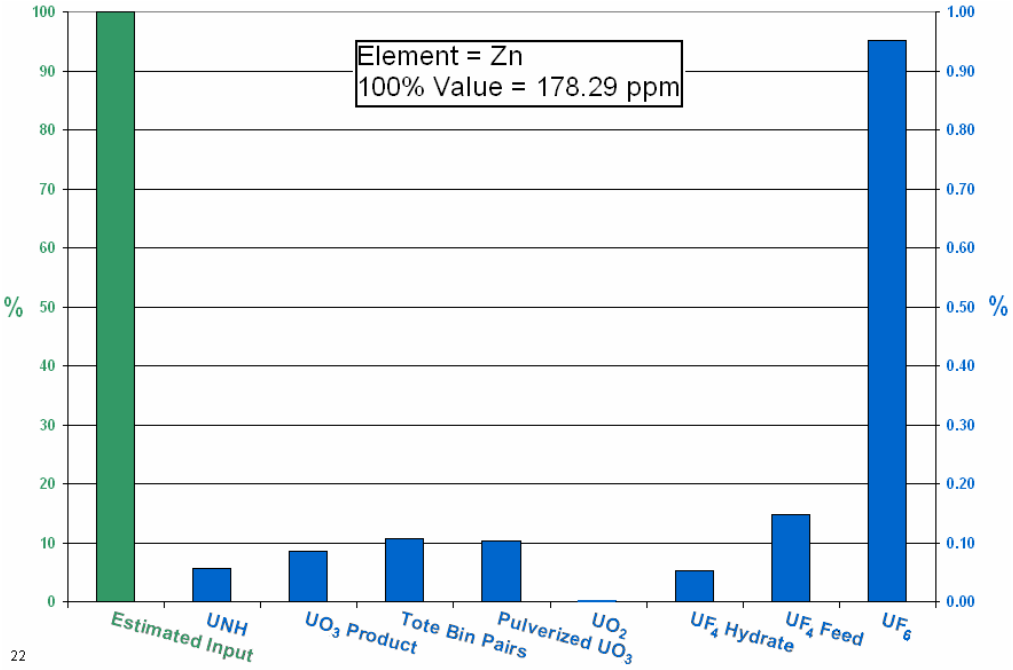


**Fig 14. Where did the Zn come from?**

## 7. Overview of Trace Element Behaviour During Conversion

The behaviour of trace elements during the conversion of UOC to $UF_6$ follows, for the most part, an expected pattern. Impure UOC is dissolved in nitric acid to produce an impure uranyl nitrate solution. The trace element content of the UOC does not change during dissolution but other chemicals are added to the dissolution process with the net effect of increasing the total trace element load. Dissolver solution undergoes solvent extraction where most (>99.5%) of the trace element load is rejected to rafinate. Purified uranyl nitrate solution is then concentrated via evaporation followed by thermal decomposition (denitration) to $UO_3$. Very little change in trace element concentration takes place during evaporation and denitration. $UO_3$ undergoes mechanical preparation for hydrogen reduction to $UO_2$. Only minor changes in trace element content occur during the preparation and reduction stages. $UO_2$ is dissolved in hydrofluoric acid to produce $UF_4$ hydrate. Trace elements are introduced via the use of recycle acid so the trace element content of the $UF_4$ hydrate is higher than for feed $UO_2$. Drying and dehydration of $UF_4$ hydrate result in only marginal changes in the trace element content of dried $UF_4$. Dried $UF_4$ is reacted with $F_2$ in flame reactors to produce $UF_6$. Fluorination creates non-volatile fluorides which drop out as ash; thus, the trace element load decreases during $UF_6$ production.

## 8. Uranium Isotopic data

Uranium Isotope Ratios

Uranium isotope ratios were reported by SAL, ITU, and QFIR, specifically the U234/U238 ratio and the U235/U238 ratio. Samples were taken from 16 lots of UOC received at Blind River from four mills. Additionally, samples were taken from 10 process steps during the conversion of UOC to $UF_6$ and some of these steps had parallel process lines. A total of 113 U isotopic ratio measurements were available. A consequence of the number of UOC sources and lots and the number and duality of process steps is that the number of samples taken for each was small, often two to three. In general the number of samples taken was insufficient for robust statistical analysis and conclusions. For ease of presentation and discussion, the isotopic ratios are given as whole numbers rounded to 5 digits.

U235/U238 Isotopic Ratios

The average value of the U235/U238 isotopic ratio for all 113 available values, regardless of the laboratory, material source, material type, or conversion activity is 72541 ($x10^{-7}$) with one standard deviation (sigma) of 17 and a relative standard deviation (RSD) of 0.02%. The RSD of 0.02% falls just below the lower value of a range of 0.03-0.05% used by SAL as the measurement uncertainty for U235/U238 isotopic ratio measurements. The data shows that the U235/U238 isotopic ratio is a constant value not influenced by mining methods, milling practices or chemical processes during conversion of UOC to $UF_6$.

U234/U238 Isotopic Ratios – UOC

There are significant differences in the U234/U238 isotopic ratios for the four uranium ore concentrates used in this study. There is a wide range of values between a maximum of 54993 and minimum of 53030, a range of 3.6%.

Taking McArthur River UOC as a reference, the isotopic ratio for ELDO UOC is 2.49% lower, for McClean Lake UOC 1.33% lower, and for Crow Butte UOC 3.13% lower. These differences are higher than the range of 0.2-0.5% used by SAL as the measurement uncertainty for U234/U238 isotopic ratio measurements. The U234/U238 isotopic ratio for all 37 values is 54246 ($x10^{-9}$) with one sigma of 686 and an RSD of 1.26%. The differences are not a result of inter-laboratory analytical differences. The differences between lab averages for UOC using the McArthur River average value as a reference are 0.22% for ITU and 0.06% for QFIR, thus at and below measurement uncertainty.

U234/U238 Isotopic Ratios – Conversion

Differences in isotopic ratios between conversion process stages are either within or below the SAL measurement uncertainty range for U234/U238 isotopic measurements (0.2-0.5%). The differences were calculated based on the difference between the isotopic ratio average of process stages and the calculated average weighted value for the U234/U238 isotopic ratio of the input to the digestion system. UOC, grit blast, and $UO_3$ recycle were included in the calculation.

Although no significant changes occur in the U234/U238 isotopic ratio during conversion stages, there appears to be an indication that changes do take place but on a very small, minor scale. It was found that the isotopic ratios of conversion stages are higher than isotopic ratios for the digestion stage calculated input. The pattern of indicated change is that from the input isotopic ratio into the digestion stage, isotopic ratio values increase through to the $UO_3$ pulverisation stage. Isotopic ratio values then drop during $UO_2$ production and drop further during the $UF_4$ and $UF_6$ production stages and appear to essentially return to very near the input to digestion values.

The pattern of isotopic ratio increase from the input baseline and then decline through $UO_2$-$UF_6$ to near baseline is unusual. The increase is unusual because there is no source of additional U234 to cause an increase in the U234/U238 ratio. Further, there is no loss mechanism during the conversion stages showing a decrease in the U234/U238 ratio to account for the decreased isotopic ratio values.

One possibility to explain the higher isotopic ratio values for the digester through pulverised $UO_3$ stages vs. the digester input values would be that the input values are too low. If the Crow Butte UOC is eliminated from the weighted average input calculation, the input average becomes essentially the same value as for the $UO_3$ isotopic ratio averages. There is no reason to suspect that the Crow Butte UOC was not included in the digester input. Even so, this would not explain the decreased values from $UO_2$ through $UF_6$. Another possibility is that the pattern is a measurement (instrument) effect of unknown cause.

## 9. Summary and Conclusions

- The trace element spectrum of feed to a mill depends on the mineralogy of the ore body plus, as applicable, the mineralogy of dilutive species accompanying the ore through mining practice or for other processing purposes. Mining and mill feed preparation practices were different for each of the four UOC sources. The associated mills each had different UOC production methods in terms

of process chemistry. These differences combine to give UOC from a mine-mill combination a unique trace element character.

- Mining and mill feed preparation practices can change over time. Mill chemical processing can also change with time.
- UOC from each source was blended at Blind River and dissolved together with additional liquid and solid process chemicals and recycle streams. Solvent extraction was effective to the degree that it was not possible to relate the trace element content of the product $UO_3$ to any contribution by any specific UOC input.
- Conversion of $UO_3$ to $UO_2$ results in only marginal changes in trace element values. The processes are mainly mechanical operations with hydrogen reduction of $UO_3$ to $UO_2$. No process chemicals (other than hydrogen) are involved.
- Conversion of $UO_2$ to hydrated and dehydrated $UF_4$ results in a notable increase (148 ppm total) in trace element values but for a limited number of elements: iron, potassium, sodium, phosphorus. The increase in iron is a result of corrosion product originating from the drum drying operation. The origin of the increased potassium, sodium, and phosphorus values is not known.
- Conversion of dehydrated $UF_4$ to $UF_6$ results in a notable decrease (171 ppm total) in trace element values but for a limited number of elements: iron, sodium, phosphorus. Thus, three of the elements responsible for increases in the trace element levels in $UF_4$ are also responsible for the decrease during $UF_4$-$UF_6$ conversion.
- Although a sophisticated analytical technique, ICP-MS can have difficulty with numerous elements. Drawing conclusions based on one set of ICP-MS trace element results from a single laboratory requires some caution.
- Before using a set of ICP-MS trace element values from any laboratory, all of the values should be carefully examined vs. available specifications or historical values and the origin and type of sample. There should be a positive answer to the question “does this result make sense?”
- The U235/U234 isotopic ratio is a constant value which is not influenced by ore source, mining methods, milling practices, or chemical conversion processes. The U235/U238 isotopic ratio could not be used to differentiate between different sources of UOC.
- The U234/U238 isotopic ratio was significantly different for each of the four uranium ore concentrates used in this study. A wide range of U234/U238 isotopic ratio values was found for each UOC source. The U234/U238 isotopic ratio has potential to differentiate between different sources of UOC.

## 10. Acknowledgements

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The IAEA’s Safeguards Analytical Laboratory
and JRC’s ITU laboratory

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## 11. References

[1] Button, P; Healey, G; Chipley, D; *Tracing Origins of uranium products, ESARDA Symposium 2011*.

# A Monte Carlo Analysis of Gas Centrifuge Enrichment Plant Process Load Cell Data

**Authors:**
**J. R. Garner**
**J. M. Whitaker**

Oak Ridge National Laboratory
P.O. Box 2008, MS6315
Oak Ridge, TN 37831-6315
E-mail: garnerjr@ornl.gov, whitakerjm@ornl.gov

***Abstract:***

*As uranium enrichment plants increase in number, capacity, and types of separative technology deployed (e.g., gas centrifuge, laser, etc.), more automated safeguards measures are needed to enable the IAEA to maintain safeguards effectiveness in a fiscally constrained environment. Monitoring load cell data can significantly increase the IAEA's ability to efficiently achieve the fundamental safeguards objective of confirming operations as declared (i.e., no undeclared activities), but care must be taken to fully protect the operator's proprietary and classified information related to operations. Staff at ORNL, LANL, JRC/ISPRA, and University of Glasgow are investigating monitoring the process load cells at feed and withdrawal (F/W) stations to strengthen international safeguards at enrichment plants. A key question that must be resolved is what is the necessary frequency of recording data from the process F/W stations? Several studies have analyzed data collected at a fixed frequency. This paper contributes to load cell process monitoring research by presenting an analysis of Monte Carlo simulations to determine the expected errors caused by low frequency sampling and its impact on material balance calculations.*

**Keywords:** GCEP Safeguards; load cell monitoring

## 1. Introduction to load cell monitoring

The International Atomic Energy Agency (IAEA) and safeguards experts are investigating monitoring the process load cells at enrichment plant feed and withdrawal (F/W) stations in an effort to strengthen IAEA safeguards [1, 2, 3, 4, 5]. As uranium enrichment plants increase in number and capacity, more automated safeguards measures are needed to enable the IAEA to maintain safeguards effectiveness while balancing fiscal constraints. Monitoring load cell data can significantly increase the IAEA's ability to efficiently achieve the fundamental safeguards objective of confirming operations as declared, but care must be taken to fully protect the operator's proprietary and classified information related to operations. One approach to protect the operator's information could be by limiting the frequency of data collection.

A key component to be resolved for effective process load cell monitoring is determining the minimum frequency of data necessary to count $UF_6$ cylinders processed, confirm full and empty cylinder weights, and calculate inventory differences. To date, several arbitrary frequencies (e.g., every 10 minutes, every 30 minutes, every 4 hours) have been used in analysis and concepts of load cell monitoring. However, drawing safeguards conclusions is driven by how long the various operational activities take at a facility.*

Uranium enrichment plants have three processes for transferring material between the process and storage cylinders: feeding, product withdrawal, and tails withdrawal. The operational activities at each

*Higher data frequencies may also contribute to procedures for building confidence in the authenticity of the data.

of these transfer stations have a slightly different cycle. Specific operation and times vary from facility to facility, but for this study the following representative activities and times are used.

At a feed station, it takes a few minutes to load a full 48Y feed cylinder into the feed station. It then takes approximately 1.5 hours to connect the hoses and purge the lines before the hatch to the feed station is closed. Once the hatch is closed, a purification process that removes light gases and other impurities takes 2–3 hours. Depending on plant conditions, the station may be placed in standby until that particular feed is needed. When the feed in that cylinder is needed, the cylinder will begin transferring $UF_6$ into the process, which takes approximately 3.5 days. When the cylinder can no longer maintain the required process pressure, other cylinders will be connected to the process. During this phase it takes about 4 hours to try to remove as much of the heel material as possible. After the heel removal process, the cylinder may sit for up to a day with the door still closed depending on the availability of a location to place the empty cylinder. After the door is opened, the cylinder cools for about 3 hours before the operators spend an hour purging and disconnecting hoses. Unless the station requires maintenance activities, the station remains empty for only a few hours before another feed cylinder is loaded in and the cycle begins again. A generic cycle is shown in Figure 1.

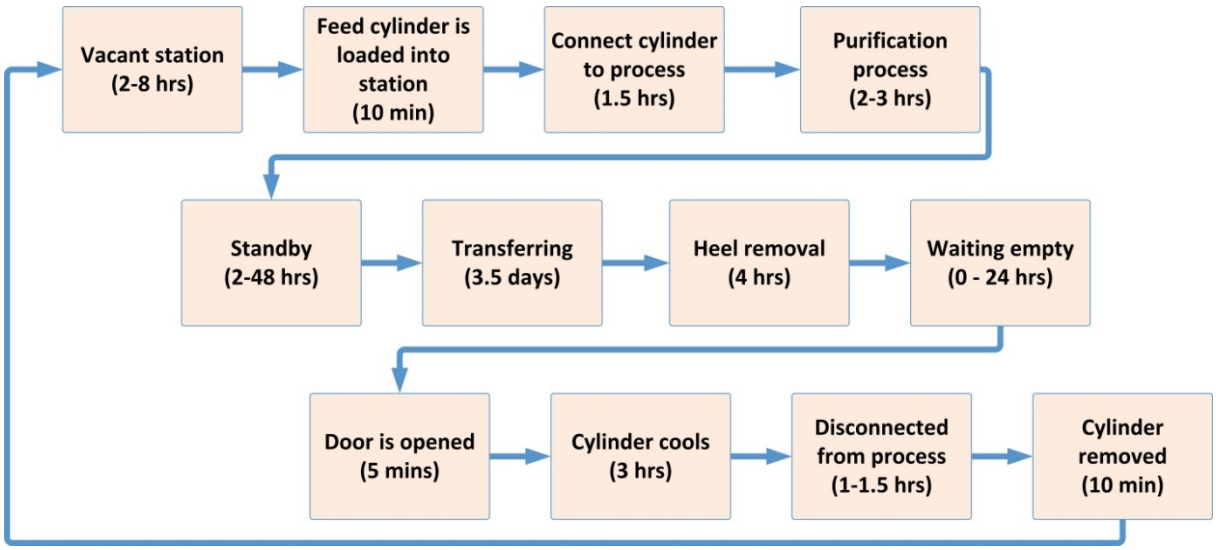



**Figure 1:** Typical feed cylinder phases

A typical F/W station at a modern plant could have four load cells connected to a junction box or summing box connected to a weighing indicator. The weighing indicator provides an excitation voltage to the load cells and converts the response signal to an electronic weight reading that is communicated to a Programmable Logic Controller (PLC) [6]. This paper assumes the IAEA is recording weight data that is split from a PLC as shown in Figure 2. Because the PLC is operator-owned equipment and each station is under the control of a central Supervisory Control and Data Acquisition System (SCADA), the PLC could forward weight and supplemental information, such as the current phase of the feed cycle. This paper assumes the weight value is part of a weigh ticket that includes the phase of the feeding activity as illustrated in Figure 1. However, if operational phase information is not available as part of a weigh ticket, the algorithm to identify the phases would likely need more data. A full discussion about how much data would be required to identify phases is beyond the scope of this paper, but the authors assume several data points for the full and empty phases would be sufficient to automatically identify these phases and the appropriate full and empty weights.

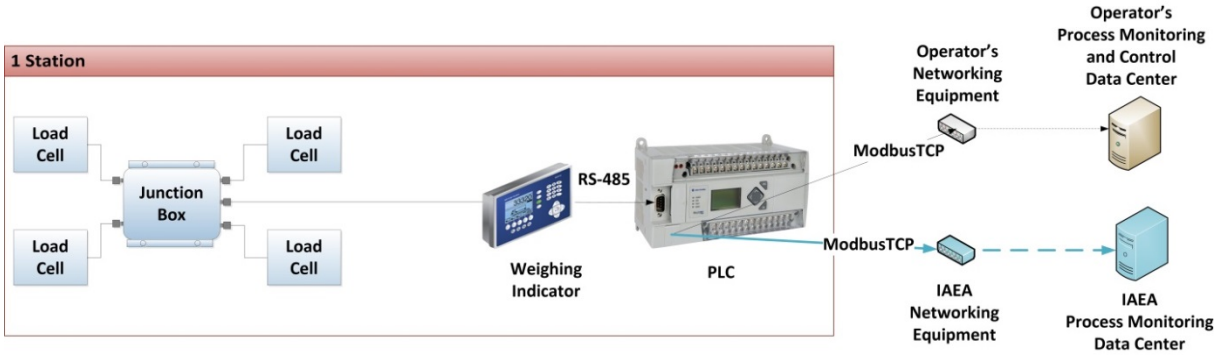



**Figure 2:** typical feed and withdrawal stations with four load cells connected to a weighing indicator, which is connected to a PLC (Signals from the operator's PLC could be shared with the IAEA and include supplemental phase information.)

The weight profile for a feed cylinder as it is fed may resemble the plot shown in Figure 3. Figure 4 shows the beginning of the feed cycle with each phase labeled, and Figure 5 shows the end of the feed cycle with each phase labeled.

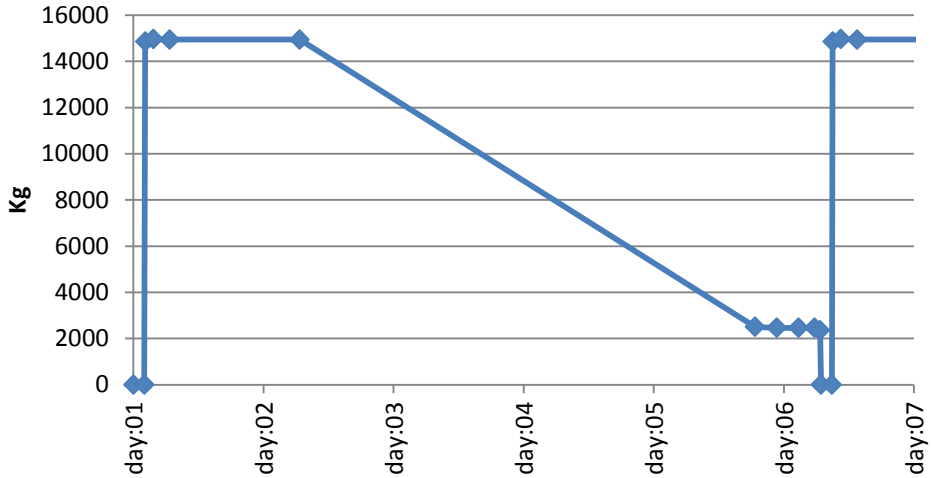



**Figure 3:** weight profile of a typical feed cycle

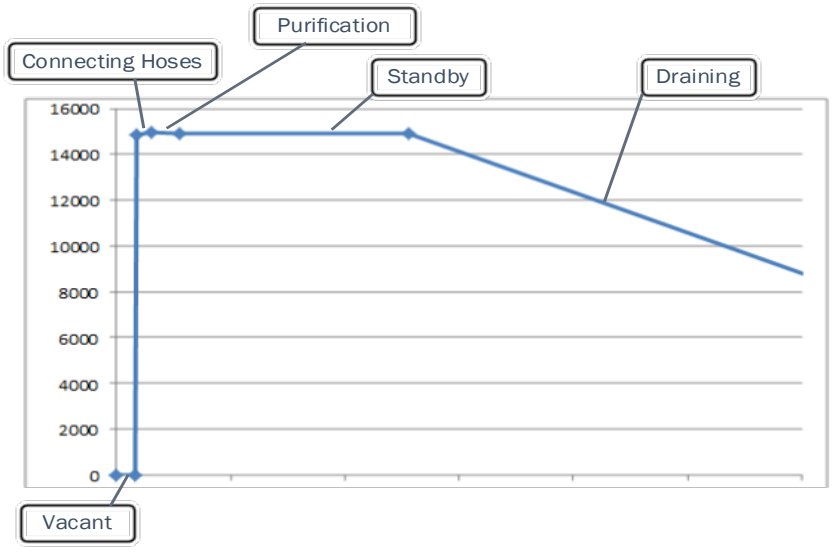



**Figure 4:** operational activities at the beginning of a feed cycle

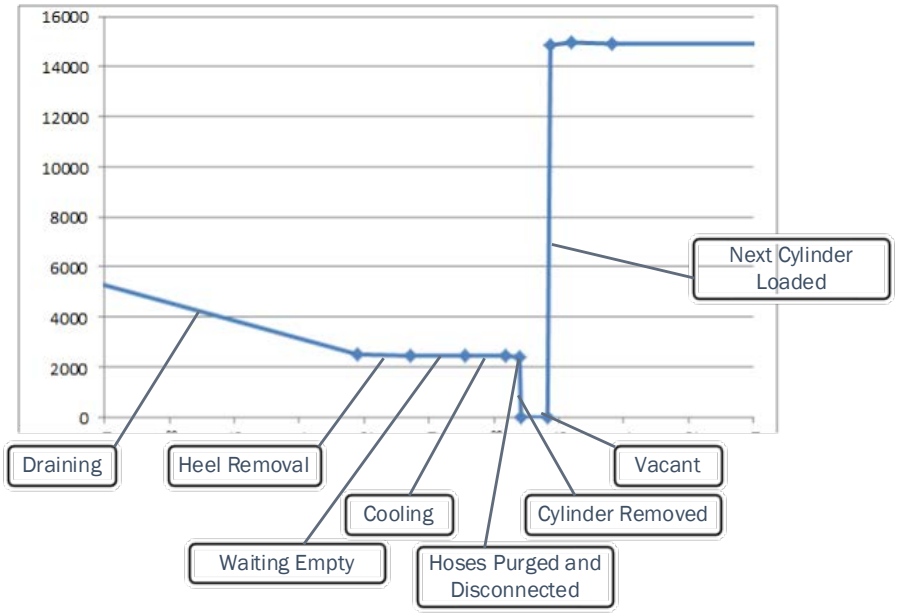



**Figure 5:** operational activities at the end of a feed cycle

From a safeguards perspective, the amount of material transferred from this feed cylinder is more important than full access to the weight time series data. However, enough time series data must be provided to accurately record the full and empty weights (and contribute to authenticating the data).

The authors have developed a mathematical model of F/W cycles to better understand the data frequency required to identify the full and empty weights during F/W cycles. The model uses the @Risk plugin for Microsoft Excel. The @Risk plugin allows users to develop models in the familiar Excel spreadsheet environment but perform Monte Carlo simulations to show many possible outcomes.

In the feed cycle example listed above, the authors generalized that it takes 1.5 hours to connect the hoses and purge the lines before the hatch to the feed station is closed, but sometimes the technicians can perform the task in 45 minutes. Other times it takes 4 hours. The PERT distribution is often used to model the duration of a task for Monte Carlo simulations [7]. As shown in Figure 6, the authors have used the PERT distribution to model the time it takes to complete a task. A PERT distribution uses a minimum time for an activity along with a maximum time, and skews the distribution between these end points using the most likely time. During a Monte Carlo simulation, @Risk samples from the distribution to generate a random duration for the time it took the technicians to connect and purge the hoses. Figure 7 shows a histogram of the random durations that @Risk selected for 20 iterations. As the number of iterations increases, the histogram more closely resembles the distribution from which the durations were sampled.

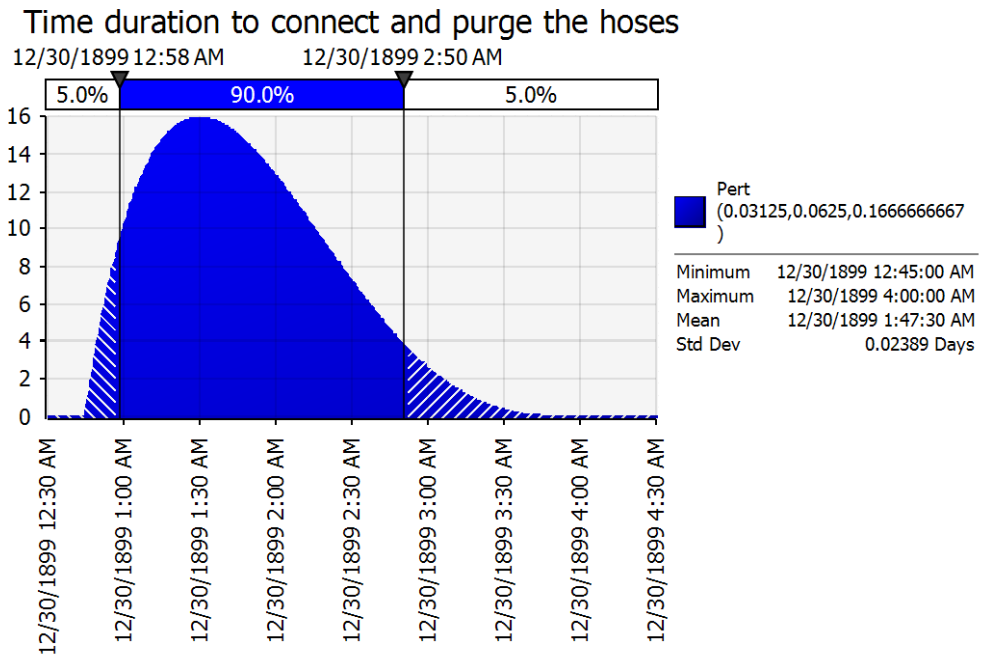



**Figure 6:** PERT distribution, which can be used to model the time it takes to complete each task (e.g., connecting and purging the hoses)

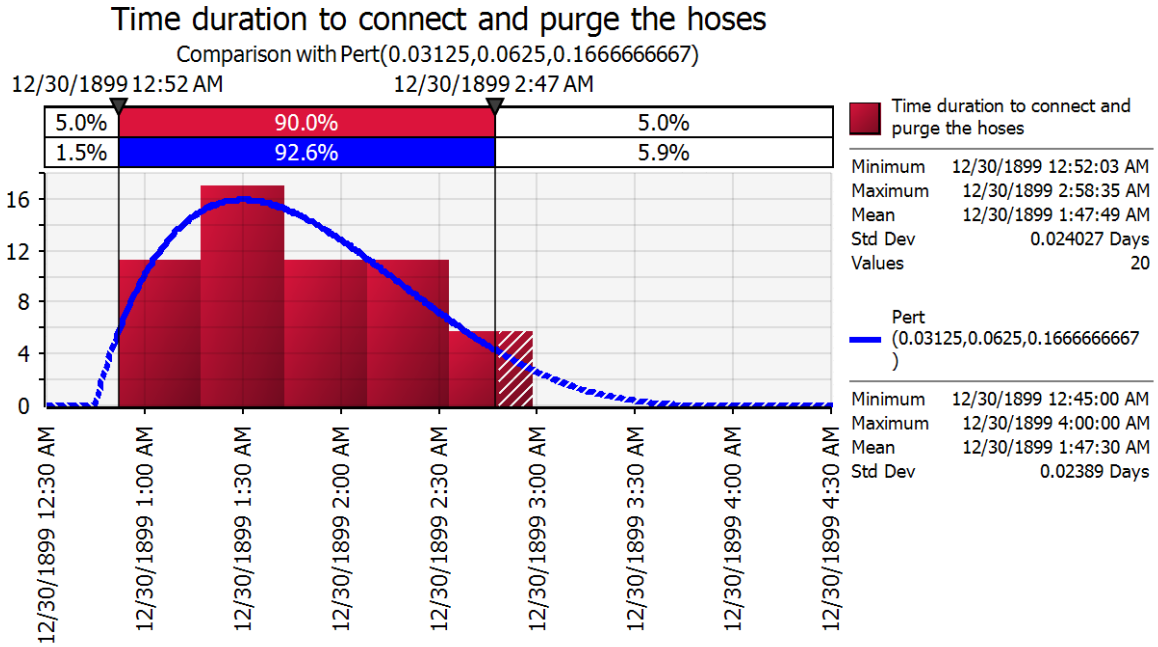



**Figure 7:** histogram showing distribution of samples after 20 iterations

This approach can be used for each phase of the feed cycle. As shown in Figure 8, the authors have used the PERT distribution to model all of the operational phases shown in Figure 1.

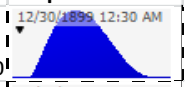
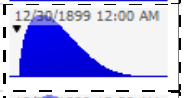
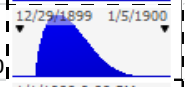
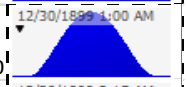
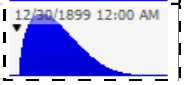
| Phase | Distribution | Min | Most likely | Max | Graph |
|---|---|---|---|---|---|
| Vacant Station | PERT | 00 days 00:03 | 00 days 01:30 | 00 days 04:00 |  |
| Feed Cylinder is loaded into station | PERT | 00 days 00:03 | 00 days 00:10 | 00 days 00:45 |  |
| Connect Cyllinder to Process | PERT | 00 days 00:45 | 00 days 01:30 | 00 days 04:00 |  |
| Purification Process | PERT | 00 days 02:00 | 00 days 03:00 | 00 days 04:00 |  |
| Standby | PERT | 00 days 01:00 | 01 days 00:00 | 05 days 00:00 |  |
| Ttransfering | PERT | 03 days 00:00 | 03 days 12:00 | 04 days 00:00 |  |
| Heel removal | PERT | 00 days 01:30 | 00 days 04:00 | 00 days 10:00 |  |
| Wainting Empty | PERT | 00 days 00:03 | 00 days 04:00 | 01 days 00:00 |  |
| Cylinder Cools | PERT | 00 days 02:20 | 00 days 03:00 | 00 days 04:00 |  |
| Disconnected from the process | PERT | 00 days 00:45 | 00 days 01:00 | 00 days 01:30 |  |
| Cylinder Removed | PERT | 00 days 00:03 | 00 days 00:10 | 00 days 00:45 |  |

**Figure 8:** distributions including parameters for feed station phases

The @Risk package can sample from these distributions for the duration of each phase to simulate complete feed cycles.

As shown in Figure 9 and Figure 10, each $UF_6$ cylinder is weighed on the accountancy scale before and after it is fed into the cascade. The authors define the full weight of a feed $UF_6$ cylinder as the gross weight of the cylinder before it is fed into the cascade, and define the “empty” weight as the weight of the feed $UF_6$ cylinder after it is fed into the cascade.

Potentially, an accurate full weight could be recorded at the feed station using a fixed frequency while the operator is connecting and purging hoses, but the authors believe the weight values collected while the hoses are being attached and purged should be considered suspect because they could be spurious due to intermittent forces from handling fittings, tightening hoses, or controlling valves. So the best full weight value would be recorded after the hatch closes but before the purification process begins. If weights are recorded on a fixed frequency, there’s no way to accurately record the full weight. In fact, the best “full” weight value would be the first weight value recorded during the purification phase.

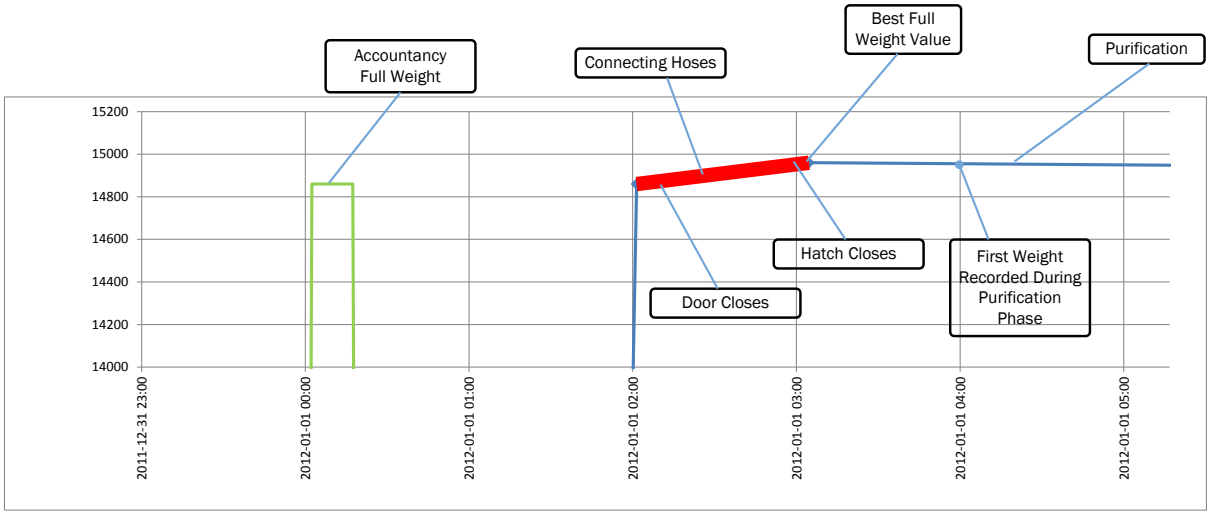



**Figure 9:** Details showing the associated full accountancy weight and the beginning of the feed cycle

In the model, the authors assumed the light gases are less than 0.5 kg.[†] As described in Table 1, if weight is reported every hour and the purification process lasts at least 3 hours, a data point during the purification process will always occur, and on average, if the scale is capable of this accuracy, the first weight reading during the purification phase will be 0.08 kg less than the actual full weight of the cylinder. Table 1 shows the likelihood of recording a weight value in the purification phase and the error between that weight and the actual full weight.

| | | 1 Data Point | |
|---|---|---|---|
| | 10,000 iterations | Purification | Mean Loss (kgs) |
| Data Point every: | 00 days 00:06 | 100.00% | 0.01 |
| | 00 days 00:30 | 100.00% | 0.04 |
| | 00 days 01:00 | 100.00% | 0.08 |

**Table 1:** Average error associated with each full feed weight from fixed frequency data collection

In our model the cylinder is held in a waiting empty phase for 0–24 hours, so an accurate empty weight can likely be recorded if weights are recorded on a fixed frequency. However, there is some likelihood that as soon as the heel removal process is complete, the operator would open the door and begin cooling the cylinder. Industry experts have indicated that after the door opens, convection currents and other temperature effects adversely affect the weight readings, so that if a weight is not recorded during the waiting empty phase, the best empty weight would be from the last weight recorded during the heel removal phase. Table 2 shows the likelihood of recording a weight value during the waiting empty phase and the average error compared to the actual empty weight when the empty process weight is recorded during the heel removal phase. These calculations assume a 50 kg heel is removed linearly. For 6 or 30 minute data collection rates, the error is less than 0.00 kg; for a 1 hour rate, the error is about 0.05 kg.

[†] A 48Y feed cylinder has a volume of 4,041 liters and a maximum shipping weight of 12,501 kg. At 20°C, $UF_6$ has a density of 5.1 g/cc, so 12,501 kg of solid $UF_6$ @ 20°C has a volume of approximate 2,451 liters, leaving 1590 liters filled with $UF_6$ gas and other gases. At 20°C the partial pressure of $UF_6$ is approximately 10.342 kPa (1.5 psia). Traditionally in the United States, cylinders have been cold burped to below 34.473 kPa (5 psia) before they are heated for feed operations. While some cylinders could have more light gases, the authors have assumed that feed cylinders would measure 34.473 kPa (5 psia) on a cold pressure check at ambient conditions assumed to be 20°C. Using the ideal gas law, this partial pressure difference of 24.131 kPa dictates that 15.72 moles of other gases would be present. Dry air has a molar mass of 28.97 g/mol, so if the gas is dry air, the gas would have a mass of 455g or approximately 0.5 kg. HF has a molar mass of 20.01 g/mol, so if the gas is HF, the mass of the gas would be even less.

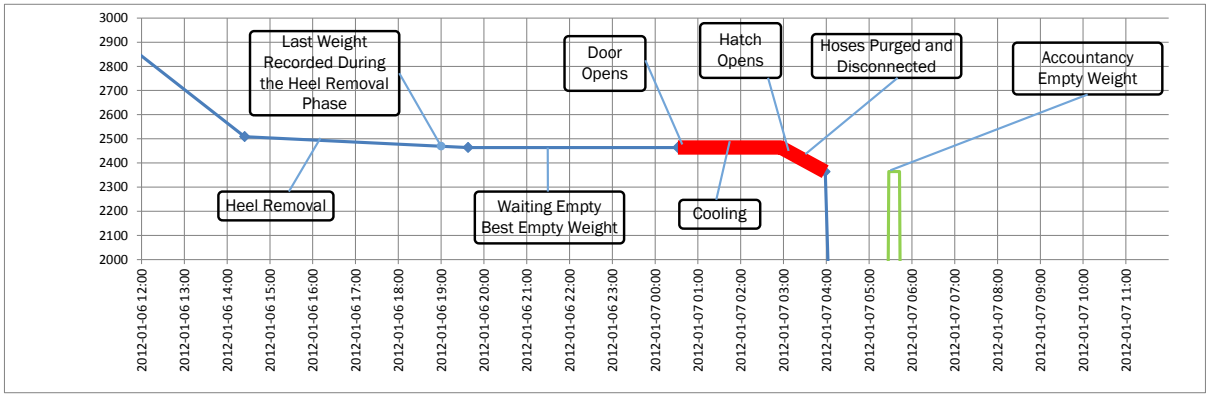



**Figure 10:** Details showing the end of the feed cycle and the associated empty accountancy weight

| | 10,000 iterations | Waiting Empty | Mean Gain (kgs) |
|---|---|---|---|
| Data Point every: | 00 days 00:06 | 99.90% | - |
| | 00 days 00:30 | 98.95% | - |
| | 00 days 01:00 | 94.40% | 0.05 |

**Table 2:** Average error associated with each empty feed weight from fixed frequency data collection

A hypothetical 4,000 tonne SWU/year GCEP could use 9258 tonnes of natural feed at 0.711% $^{235}$U to enrich 1552 tonnes of product $UF_6$ to 3.0% $^{235}$U with 7706 tonnes of tails depleted to 0.25% $^{235}$U. If we assume the plant uses 48Y cylinders for feed and tails, and 30B cylinders for product to 95% of their full capacity, the plant would require about 780 48Y feed cylinders, 720 30B product cylinders, and 649 48Y tails cylinders (Figure 11).

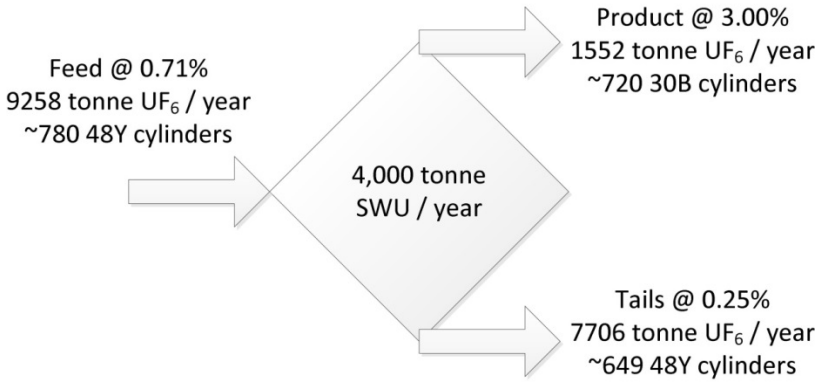



**Figure 11:** Cylinder requirements at a 4,000 tonne SWU/year facility (The facility could use 780 48Y cylinders to produce 720 30B product cylinders at 3%, and 649 48Y tails cylinders at 0.25%.)

The error from using weight values after the purification phase started was seen on average as 0.08 kg less than the accountancy full weight. For a simulation consisting of 780 feed cycles, this totaled about 66 kg. The error from using a weight value during the heel removal phase was seen on average as 0.05 kg more than the accountancy empty weight, which totaled about 35 kg for 780 iterations. So for 780 iterations, or approximately the number of feed cycles annually at our hypothetical 4,000 tonne SWU/year GCEP, if uncertainties are ignored, the inventory difference calculated on the accountancy scale data would show that 31 kg more of natural $UF_6$ was fed to the process than the F/W station load cells reported.

## 2. Comparison with uncertainty values

Many sources report that a load cell typically provides 0.02% to 0.03% full scale accuracy [8, 9]. Typically, large loads like $UF_6$ cylinders are weighed using weighing systems that consist of several

load cells. When several load cells are used together, each load cell only needs to be rated for a portion of the total rated capacity of the weighing system. As a worst-case assumption, we can assume that the F/W stations use four load cells each rated to the full capacity of a 48Y or 30B cylinder. So, if a full 48Y feed or tails cylinder weighs about 14,860 kg, as a worst case we could assume each feed or tails withdrawal station uses four 15,000 kg rated load cells. And if a full 30B cylinder weighs about 2912 kg, as a worst case we could assume each product station uses four 3,000 kg rated load cells. A junction box is used to add the signals from the four load cells, so the total weight can be thought of as the addition of the weights reported from each load cell. If the four load cells report weights $w_1$, $w_2$, $w_3$, and $w_4$ with random uncertainties $\sigma_{w1}$, $\sigma_{w2}$, $\sigma_{w3}$, and $\sigma_{w4}$, respectively, the total weight of a cylinder in a F/W station at a given time is given by **Equation 1**. Because the uncorrelated and random errors for equations with simple addition or subtraction are equal to the square root of the sum of the squares of the individual errors, the uncertainty associated with the total weight measurement is given by Equation 2.

**Equation 1: Total weight from four load cells**

$$w_{total} = w_1 + w_2 + w_3 + w_4$$

**Equation 2: Uncertainty associated with a single weight measurement**

$$\sigma_{w_{total}} = \sqrt{{\sigma_{w1}}^2 + {\sigma_{w2}}^2 + {\sigma_{w3}}^2 + {\sigma_{w4}}^2}$$

In the product stations, if four 3000 kg load cells are used with 0.03% full scale accuracy to weigh 30B cylinders, the weight reported from each load cell would have a standard deviation of 0.9 kg. Applying Equation 3, each product weight value would have 1.8 kg of uncertainty.

**Equation 3: Uncertainty from each product cylinder weight value**

$$\sigma_{48Y_{UF_6}} = \sqrt{0.9^2 + 0.9^2 + 0.9^2 + 0.9^2} = 1.8$$

Similarly, if the 48Y cylinders are weighed using four 15,000 kg load cells with 0.03% full scale accuracy, the weight reported from each load cell would have a standard deviation of 4.5 kg. As shown in Equation 4, the feed and tails 48Y cylinder weight values would have 9 kg of uncertainty.

**Equation 4: Uncertainty from each feed and tails cylinder weight value**

$$\sigma_{48Y_{UF_6}} = \sqrt{4.5^2 + 4.5^2 + 4.5^2 + 4.5^2} = 9$$

For a bulk handling facility, we would expect that a mass balance of the process area would be non-zero. As shown in Equation 5, the difference between the mass of the material fed to the process and withdrawn is called the inventory difference. The IAEA regards inventory differences as Material Unaccounted For (MUF).

**Equation 5: Uncertainty from each product cylinder weight value**

$$ID = \sum \{F - (P + T)\}$$

The amount of material fed to the cascade is the difference between the full feed cylinder weight and the empty feed cylinder weight. Similarly, the material withdrawn from the cascade is the difference between the empty withdrawal cylinders and the full withdrawal cylinders. So, the mass balance for the hypothetical 4,000 tonne facility that handles approximately 780 feed cylinders, 720 product cylinders, and 649 tails cylinders could be calculated as shown in Equation 6. Because this is simple addition and subtraction, the uncertainty associated with this calculation can be calculated using the square root sum of the squares method discussed above. Therefore, the uncertainty in the annual mass balance from the process scales in terms of $UF_6$ is shown in Equation 7.

**Equation 6: Mass balance**

$$MUF = \sum (F_{1_{Full}} - F_{1_{Empty}}) + \left(F_{2_{Full}} - F_{2_{Empty}}\right) + \cdots + \left(F_{780_{Full}} - F_{780_{Empty}}\right) - \left\{ \begin{array}{l} \left(P_{1_{Full}} - P_{1_{Empty}}\right) + \left(P_{2_{Full}} - P_{2_{Empty}}\right) + \cdots + \left(P_{720_{Full}} - P_{720_{Empty}}\right) + \\ \left(T_{1_{Full}} - T_{1_{Empty}}\right) + \left(T_{2_{Full}} - T_{2_{Empty}}\right) + \cdots + \left(T_{649_{Full}} - T_{649_{Empty}}\right) \end{array} \right\}$$

**Equation 7: Uncertainty associated with an annual mass balance**

$$\sigma_{UF_6} = \sqrt{9^2 + 9^2 + \cdots + 1.8^2 + 1.8^2 + \cdots + 9^2 + 9^2} = \sqrt{2 * 780 * (9^2) + 2 * 720 * (1.8^2) + 2 * 649 * (9^2)} = 486\ kgUF_6$$

A non-zero MUF is expected for several reasons, including due to the uncertainties associated with all of the feed product and tail cylinder weight measurements. The MUF can become a concern when it is larger than a significant quantity (SQ). The IAEA defines one significant quantity of indirect use material as 75 kg $^{235}$U, 10 tonnes of natural U, or 20 tonnes of depleted U [10]. Equation 8 and Equation 9 show the uncertainty in the weight values from each 48Y and 30B in terms of kilograms of U. Equation 10, Equation 11, and Equation 12 show the uncertainty in the weight values from each feed, product, and tails cylinder in terms of kilograms of $^{235}$U.

**Equation 8: Uncertainty in weight of U from each 48Y feed and tails cylinder weight value**

$$\sigma_{48Y_U} = 9\ kg\ UF_6 * 0.676\ \frac{U}{UF_6} = 6.084\ kg\ U$$

**Equation 9: Uncertainty in weight of U from each 30B product cylinder weight value**

$$\sigma_{30B_U} = 1.8\ kg\ UF_6 * 0.676\ \frac{U}{UF_6} = 1.21\ kg\ U$$

**Equation 10: Uncertainty in weight of $^{235}$U from each 48Y feed weight value**

$$\sigma_{48Y_{^{235}U}} = 9\ kg\ UF_6 * 0.676\ \frac{U}{UF_6} * 0.00711\ ^{235}U = 0.0433\ kg\ ^{235}U$$

**Equation 11: Uncertainty in weight of $^{235}$U from each 30B 3% product weight value**

$$\sigma_{30B_{^{235}U}} = 1.8\ kg\ UF_6 * 0.676\ \frac{U}{UF_6} * 0.03\ ^{235}U = 0.0365\ kg\ ^{235}U$$

**Equation 12: Uncertainty in weight of $^{235}$U from each 48Y 0.25% tails weight value**

$$\sigma_{48Y_{^{235}U}} = 9\ kg\ UF_6 * 0.676\ \frac{U}{UF_6} * 0.0025\ ^{235}U = 0.015\ kg\ ^{235}U$$

The uncertainty in the annual mass balance from the process scales in terms of U is shown in Equation **13**, and the uncertainty in the annual mass balance from the process scales in terms of $^{235}$U is shown in Equation **14**.

**Equation 13: Uncertainty associated with an annual mass balance in terms of kg U**

$$\sigma_{UF_6} = \sqrt{6.084^2 + 6.084^2 + \cdots + 1.21^2 + 1.21^2 + \cdots + 6.084^2 + 6.084^2} = \sqrt{2 * 780 * (6.084^2) + 2 * 720 * (1.21^2) + 2 * 649 * (6.084^2)} = 328\ kgU$$

**Equation 14: Uncertainty associated with an annual mass balance in terms of kg $^{235}U$**

$$\sigma_{UF_6} = \sqrt{0.0433^2 + 0.0433^2 + \cdots + 0.0365^2 + 0.0365^2 + \cdots + 0.015^2 + 0.015^2}$$
$$= \sqrt{2 * 780 * (0.433^2) + 2 * 720 * (0.0365^2) + 2 * 649 * (0.015^2)}$$
$$= 2.27\ kg^{235}U$$

Table 3 computes the one sigma and three sigma uncertainties in terms of total $UF_6$ and total $^{235}U$ for the given facility assumptions if the facility has different uncertainties for each feed, product, or tails weight reading. In Table 3, the results from the mass balance one sigma uncertainty calculation (Equation 7) are highlighted in blue, and the results from
Equation **14** can be found just to the right highlighted in green. As highlighted in yellow in Table 3, for the reference facility with uncertainties as large as 95 kg for 48Y cylinders and 20 kg for 30B cylinders (more than an order of magnitude larger than we expect), the three sigma uncertainties for $^{235}U$ could be less than one significant quantity.

| | | Standard Deviation | | | | | σ | | 3σ | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | kg UF6 | kg U | U235 | kg U235 | # weighings | total UF6 | total U235 | total UF6 | total U235 |
| Feed | 48Y | 9 | 6.084 | 0.71% | 0.043257 | 780 | | | | |
| Product | 30B | 1.8 | 1.2168 | 3% | 0.036504 | 720 | 486 | 2.27 | 1458 | 6.80 |
| Tails | 48Y | 9 | 6.084 | 0.25% | 0.01521 | 649 | | | | |
| | | | | | | | | | | |
| | | Standard Deviation | | | | | | | | |
| | | kg UF6 | kg U | U235 | kg U235 | # weighings | total UF6 | total U235 | total UF6 | total U235 |
| Feed | 48Y | 20 | 13.52 | 0.71% | 0.096127 | 780 | | | | |
| Product | 30B | 5 | 3.38 | 3% | 0.1014 | 720 | 1086 | 5.54 | 3258 | 16.6 |
| Tails | 48Y | 20 | 13.52 | 0.25% | 0.0338 | 649 | | | | |
| | | | | | | | | | | |
| | | Standard Deviation | | | | | | | | |
| | | kg UF6 | kg U | U235 | kg U235 | # weighings | total UF6 | total U235 | total UF6 | total U235 |
| Feed | 48Y | 95 | 64.22 | 0.71% | 0.456604 | 780 | | | | |
| Product | 30B | 20 | 13.52 | 3% | 0.4056 | 720 | 5135 | 24 | 15405 | 73.2 |
| Tails | 48Y | 95 | 64.22 | 0.25% | 0.16055 | 649 | | | | |
| | | | | | | | | | | |
| | | Standard Deviation | | | | | | | | |
| | | kg UF6 | kg U | U235 | kg U235 | # weighings | total UF6 | total U235 | total UF6 | total U235 |
| Feed | 48Y | 100 | 67.6 | 0.71% | 0.480636 | 780 | | | | |
| Product | 30B | 50 | 33.8 | 3% | 1.014 | 720 | 5673 | 43.3 | 17018 | 130 |
| Tails | 48Y | 100 | 67.6 | 0.25% | 0.169 | 649 | | | | |
| | | | | | | | | | | |
| | | Standard Deviation | | | | | | | | |
| | | kg UF6 | kg U | U235 | kg U235 | # weighings | total UF6 | total U235 | total UF6 | total U235 |
| Feed | 48Y | 250 | 169 | 0.71% | 1.20159 | 780 | | | | |
| Product | 30B | 70 | 47.32 | 3% | 1.4196 | 720 | 13626 | 73.4 | 40879 | 220 |
| Tails | 48Y | 250 | 169 | 0.25% | 0.4225 | 649 | | | | |

**Table 3:** How different uncertainties affect total uncertainty for the year

## Conclusion

The weakness of the current GCEP safeguards approach is that it isn't well suited for detecting undeclared $UF_6$ feed. This paper demonstrates a potential solution through the use of hourly, unattended load cell data. The frequency of the data transmissions will be a key factor in determining how to protect the operators' proprietary information. Load cell monitoring could be one tool in the IAEA's tool box and may not necessarily be deployed/utilized at all GCEPs, and in all cases would require host country/operator consent. Load cell data could be integrated with accountancy scale data, OLEM, UCVS, etc. to make safeguards more effective and efficient.

The authors must caution that this initial analysis is heavily dependent on how the plant is operated and does not analyze the impact to the MUF calculation as a result of an operator trying to divert into MUF.

If the distributions used to model the duration of each phase are representative and reasonable, this analysis shows that for a 4,000 tonne SWU/year facility with weight data provided every hour, the systematic errors from feed station data will be less than the measurement uncertainty.

This analysis shows that load cell data collected every hour could be considered a continuity-of-knowledge tool to be used in conjunction with accountancy scale measurements. As presented in Equation 4, the uncertainty associated with each process load cell weight is actually larger than the systematic error due to recording a weight during the purification process. Instead of using the process load cell data for material balance, the process load cells offer tremendous continuity of knowledge to cover the excess production scenario for undeclared feed.

At larger plants, the number of cylinders would likely increase, which would also increase the uncertainties in material balance calculations from process load cell data. To counteract this, each enrichment unit or module could be treated as a separate MBA, or the material balance could be calculated more frequently.

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# Safeguards-, Safety- and Security-by-Design after Fukushima

Joseph F. Pilat and Kory W. Budlong Sylvester
Los Alamos National Laboratory[1]

**Abstract**

Before Fukushima, the prospect of global nuclear expansion appeared to be defining the future nuclear landscape as a result of increasing energy demand, growing concerns over energy security and climate change and other emerging issues. It was believed that the prospects for dramatic growth in nuclear power would depend not only on economic competitiveness, but also on safety, security and proliferation resistance. Whatever the long-term impact of Fukushima on global interest in nuclear power, the pre-accident rationale for nuclear power remains, and it still appears at present that nuclear power will expand, although at a slower rate and less widely, than might have been the case before the accident. Significant new construction will be needed simply to replace aging infrastructure. However, even limited growth scenarios will depend on, inter alia, whether concerns about safety, security and proliferation and terrorism risks can be addressed.

Ensuring safeguards, safety and security—the so called "3S"—is critical to the future of nuclear power. Current efforts to promote proliferation resistance may offer one path to dealing with 3S. Although proliferation resistance has often been oversold, there is a promising path ahead with reactors and the fuel cycles they are imbedded in designed to minimize proliferation risks coupled with effective safeguards and other non-proliferation measures. The novel approach involves new technologies, facility designs and institutional arrangements that embody cost-effective means to safeguard material or, where possible, to allow the refitting of old facilities to make them more "safeguardable." The objective of safeguardability or safeguards-by-design, is to improve the application of safeguards by, for example, reducing or eliminating diversion or misuse pathways or increasing the prospects for detection by facilitating verification. Safeguards-by-design could be complemented by a specific "security-by-design" effort, and there have been discussions about such an integrated approach. Safety by design has been to date more of an industry priority. Integrating it with safeguards and security has not yet happened, but could be a useful adjunct to ongoing efforts to realize safeguards and security design objectives. If these efforts are successful, and they may not be, they could in principle strengthen all three objectives by building on common systems and protocols, while minimizing tensions, for example, between safety and security.

However attractive, the feasibility, costs, operational impacts and other possible effects of such approaches have not yet been demonstrated and will need to be analyzed further. This paper begins the analysis by placing 3S issues in the context of the nuclear governance debate. It then examines the prospects for safeguards-, security- and safety- by-design, identifies the issues raised by efforts to integrate them and demonstrate a "business case" for doing so, and offers thoughts on a 3S technology R&D program.

**Keywords**: Nuclear Energy; Safeguards; Security; Safety

## Introduction

Before Fukushima, the prospect of global nuclear expansion appeared to be defining the future nuclear landscape as a result of increasing energy demand, growing concerns over energy security and climate change and other emerging issues. It was believed that the prospects for dramatic growth in nuclear power would depend not only on economic competitiveness, but also on safety, security and proliferation resistance. Whatever the long-term impact of Fukushima on global interest in nuclear power, the pre-accident rationale for nuclear power remains, and it still appears at present that nuclear power will expand, although at a slower rate and less widely, than might have been the case before the accident. Significant new construction will be needed simply to replace aging

[1] The views expressed are the authors' own and not those of the Los Alamos National Laboratory, the National Nuclear Security Administration, the Department of Energy or any other agency.

infrastructure. However, even limited growth scenarios will depend on, inter alia, whether concerns about safety, security and proliferation and terrorism risks can be addressed.

Ensuring safeguards, safety and security—the so called "3S"—is critical to the future of nuclear power. This paper begins the analysis by placing 3S issues in the context of the nuclear governance debate. It then examines the prospects for safeguards-, security- and safety- by-design, identifies the issues raised by efforts to integrate them and demonstrate a "business case" for doing so, and offers thoughts on a 3S technology R&D program.

## Nuclear Governance and 3S

The framework for global nuclear governance is a patchwork, with widely differing rights and obligations and significant gaps. With the International Atomic Energy Agency's (IAEA's) role, reinforced by the Treaty on the Nonproliferation of Nuclear Weapons (NPT) and other treaty obligations, the international nuclear nonproliferation regime is the most well developed mechanism for verifying compliance with safeguards. However, it is increasingly being challenged by noncompliance and other problems. The nonproliferation regime has some residual impact on nuclear security and there have been efforts to extend its provisions to this arena, notably through UN Security Council Resolution 1540. Despite this, there is nothing comparable to the nonproliferation regime in the nuclear security or safety realms, which have been historically (and are presently) seen as domestic responsibilities. In these areas, there is no international entity with the ability to regulate, but several relevant industrial groups are promoting international standards and best practices, i.e., the World Association of Nuclear Operators (WANO) and the World Institute for Nuclear Security (WINS).

Although global institution building for good nuclear governance should be the objective, it is a longer term goal at best and unlikely to be accepted fully any time soon. Other more limited efforts are being pursued in the near term. As noted, in the nuclear safety and security arenas, agreements and arrangements have been reached, and best practices, standards and norms are developing and capabilities are growing in the IAEA and elsewhere. Regional institutions and other fora can also promote cooperation on safety, security and safeguards.

Further, better integrating safeguards, security and safety is possible and desirable, albeit difficult to realize. A near- to medium-term approach should build upon current efforts, including those initiated after Fukushima, and strengthen them through expanding the role of the IAEA, building on and expanding ongoing proliferation resistance and physical protection activities and furthering enabling technology R&D programs.

## Expanding the Role of the IAEA

Strengthening, pursuing universal adherence to and improving implementation of all safety, security and nonproliferation arrangements, agreements and instruments will be important. However, it is increasingly being argued that a global response to the challenges of integrating safety, security and safeguards in the post-Fukushima environment should begin with the IAEA. Although there are opponents of expanding the IAEA's role, the Agency has a mandate to promote nuclear power and administer safeguards, and it also has explicit and growing roles in nuclear safety and security.

An effective, strengthened international safeguards system, with a strong focus on verification and searching for undeclared nuclear materials and activities, is essential to provide confidence that shared nuclear technologies and expertise, as well as nuclear materials themselves, are not being diverted to nuclear-weapon programs. The state-level concept being championed by the IAEA is especially important and should offer a level of transparency and openness that is unprecedented and could have impacts well beyond safeguards themselves.

In addition to efforts to improve the effectiveness and efficiency of safeguards, on the basis of current plans and other factors, the IAEA will be inspecting third and fourth generation nuclear reactors and will need to prepare for enrichment plants using new technologies, pyroprocessing plants and other new fuel cycle facilities, along with multinational fuel cycle centers and international fuel banks.

Moreover, regardless of growth, increasingly, old plants will be decommissioned; and growing stocks of plutonium and spent fuel (including defense material declared excess by nuclear-weapon States) will have to be safeguarded.

Greater inspection responsibilities, especially if nuclear energy expands, and the introduction of new technologies and fuel cycles, will require the Agency to ensure that safeguards are credible and efficient, and that it has the technical, human and financial resources to meet these challenges. Efforts to strengthen safeguards, ensure the necessary R&D, develop a new generation of inspectors and analysts and maintain and improve infrastructure are being pursued by the Agency and Member States (e.g., the US Next Generation Safeguards Initiative, or NGSI) and will need an enduring and sustainable commitment.

While the IAEA is strengthening its safeguards, it is also being requested to expand its role in the safety and security arenas. The IAEA has had a longstanding role in safety and security, which has been evolving in response to global challenges. The accident at Chernobyl led to an expansion of the IAEA's role and capabilities in safety. Despite the Agency's initial difficulties in dealing with Fukushima, it was able to play a positive role. As a consequence, many if not all Member States have called for a further expansion in the safety area. The adoption of the Action Plan on Nuclear safety in 2011 was a promising first step.

As for security, the increasing concerns about radiological and nuclear terrorism following the attacks of September 11, 2001, led to an ongoing effort to address the nuclear terrorism threat in the IAEA. The IAEA Board of Governors approved in March 2002 a program to strengthen nuclear and radiological material security around the globe. The IAEA's program provides technical advice, legislative and regulatory assistance, training, equipment upgrades, peer reviews, etc. In addition to working with states, the IAEA coordinates with other UN agencies and bodies and has, importantly, helped in an effort to create or strengthen old legal instruments, most notably, the successful IAEA-led effort to amend the Convention on the Physical Protection of Nuclear Material, which extends the scope of the Convention to cover, inter alia, the physical protection of nuclear material used for peaceful purposes, in domestic use, storage and transport and the physical protection of nuclear material and the protection of peaceful nuclear facilities against sabotage.

There are good reasons why the Agency should be expanding its role in these areas. There is a growing advocacy for it to do so, as seen in statements by key states at the IAEA General Conference, the UN General Assembly and other nuclear fora after Fukushima. However, there are also challenges to pursuing these growing roles, and some institutional risks associated with greater Agency involvement in safety and security, where national responsibilities remain central. There also will be concerns about the Agency's ability to obtain the required resources and to pursue these new missions without undermining its traditional nonproliferation mission, especially since it is being requested to expand its role in arms control and disarmament.[2] In all, the Agency will move ahead in the safety and security arenas, and needs to ensure it has the capabilities, resources and Member State support necessary to be successful.

**Building on Proliferation Resistance and Physical Protection Efforts**

Initiatives to address new and emerging proliferation and terrorism risks through threat reduction, detection and interdiction have been put forward over the last 20 years. These include Cooperative Threat Reduction programs, the Proliferation Security Initiative, the Global Initiative to Combat Nuclear Terrorism, UNSC Resolution 1540, the Convention on the Suppression of Nuclear Terrorism and the amendments to the Convention on the Physical Protection of Nuclear Material. There are also several ideas that are being considered, or rather reconsidered, today to address proliferation and

[2] The Agency's role in arms control and disarmament—although envisioned in the Atoms for Peace speech that put forward the idea of the IAEA, embodied in the IAEA Statute and intermittently pursued in recent decades—has not been at the forefront of thinking about the IAEA's future. The fissile material cutoff treaty (FMCT) and growing interest in disarmament could change this situation. If the Agency expands its role in the arms control and disarmament arena, it will be necessary to ensure the Agency has adequate resources. The Agency will also have to overcome the impact of potential compliance disputes on its credibility and effectiveness, the possible inadvertent spread of classified or proliferation-sensitive information and other issues.

terrorism risks, including fuel banks, cradle-to-grave fuel services, multinational approaches and proliferation resistance.

While all of these efforts will need to continue if nuclear proliferation and terrorism risks are to be successfully mitigated, building on current work on proliferation resistance[3] may offer an attractive path to integrating safety, security and safeguards.

Although proliferation resistance has often been oversold, and has had limited results to date, there is a promising path ahead with reactors and other facilities designed to minimize proliferation risks coupled with effective safeguards and other non-proliferation measures that could have utility for states as well as for non-state actors. The novel approach involves new technologies and facility designs that embody cost-effective means to safeguard material or, where possible, to allow the refitting of old facilities to make them more "safeguardable." The objective of safeguardability or safeguards-by-design, is to improve the application of safeguards by, for example, reducing or eliminating diversion or misuse pathways or increasing the prospects for detection along pathways by facilitating verification through, for example, technologies and approaches involving greater physical access as well as increased process and operating information.

Safeguards by design should be complemented by a specific "security by design" effort, and there have been discussions about such an integrated approach. It appears that fuel cycle design, facility layouts and protective measures can reduce or eliminate the feasibility of certain theft scenarios and make their success extremely unlikely. Such design measures could include steps to track and locate nuclear material accurately and cheaply throughout the cycle; to eliminate scenarios that are costly to address with external measures (i.e., additional guns, gates and guards); to make a facility inaccessible to outsiders (and to insiders for unauthorized actions); and to increase the role of "passive" security features (e.g., material matrices that are self-protecting).

Safety by design has been to date more of a priority for industry and academia. Integrating it with safeguards and security has not yet happened, but could be a useful adjunct to ongoing efforts to realize safeguards and security design objectives.

**3S by Design**

In the post-Fukushima environment, there may be incentives for so integrating safety with safeguards and security. However, in addition to optimizing synergies, addressing tensions between the three objectives will be necessary. These issues are significant, and are made more complex by the different (and at times competing) interests of designers who need to meet performance requirements, ensure efficiency and limit costs; safety regulators who must license facilities, monitor operations and verify compliance with regulations; security regulators, who need to understand vulnerabilities, ensure facilities and operations are covered by a viable security plan and monitor implementation to ensure procedures are followed; and the IAEA, which needs to develop a safeguards approach, gather necessary data and verify declarations.

In examining issues such as access, transparency and operational flexibility, each actor in the 3S process will see each issue from their unique perspective, either in agreement or at odds with one another. (See Table 1.) At the operational level, the focus for facility designers will be on the provision of sufficient access to support efficient operations and maintenance but not more. Safety and

[3] Although there has been some controversy and continued debate over the meaning of proliferation resistance, the International Atomic Energy Agency (IAEA) developed a widely accepted definition that remains interested in the technological elements of proliferation resistance that go back to the Acheson-Lilienthal report but also looks explicitly at institutional elements. According to the IAEA's definition: "Proliferation resistance is that characteristic of the nuclear energy system that impedes the diversion or undeclared production of nuclear materials, or misuse of technology, by the host state in order to acquire nuclear weapons or other nuclear explosive devices." Intrinsic proliferation resistant features, according to the IAEA, "are those features that result from the technical design of nuclear energy systems, including those that facilitate the implementation of the extrinsic measures." As for extrinsic measures, the IAEA's definition states: "Extrinsic proliferation resistance features are those features that result from the decisions and undertakings of states related to nuclear energy system."

safeguards will seek ever higher access to serve their respective interests. This access can pose a potential problem for security, forcing limitations. Security interests generally favor restricting access in the design to make it inherently difficult for unauthorized personnel to enter a facility.

However, this interplay is in reality more complex and context dependent. For example, from a regulatory perspective, it is always desirable to have access to monitor safe operations. Nevertheless, it will be important to design in restrictions to access (protections) for workers and inspectors alike when radiation exposure is too high or dangerous operations are underway. From a safeguards perspective, the Agency would like to have ready access to any part of the plant with little or no notice. However, in looking at safeguardability, from a containment/surveillance perspective, a static facility is easier to safeguard than a dynamic one. In this case, from a design perspective, they would like to limit regular access (e.g., fewer penetrations to monitor, surveillance easier to assess, etc.).

Taken as a measure of openness, external regulators will always be in favor of transparency to the degree, at least, that it supports their efforts. However, the transparency desirable for safeguards and safety is not desired for security, where opacity may be an integral part of a security strategy which would work against such measures. Operational flexibility reinforces the view that it is often difficult to draw sharp lines. Operational flexibility is typically desirable from the operator's perspective, depending on cost. From a safeguards perspective, this would make it more difficult for the IAEA to monitor the facility. To the extent it adds to complexity, it becomes an issue for security. Moreover, if it adds to risk, it will be a safety problem.

The levels of international authority, norms and standards, as well as the manner in which industry best practices are pursued differ significantly. Best practices are easier to promote in the area of safety where there is greater consensus around methods of assessing performance and there are fewer sensitivities involved.

| | Safety | Security | Safeguards |
|---|---|---|---|
| ***Operational*** | | | |
| **Access** | Yes | No | Yes |
| **Transparency** | Yes | No | Yes |
| **Flexibility** | Maybe | No | No |
| **Remote Siting** | Yes | Yes | N/A |
| **Self-protection** | N/A | Yes | Maybe |
| ***Institutional*** | | | |
| **International Authority** | No | No | Yes |
| **International Norms/Standards** | Yes | Limited | Yes |
| **Industry Best Practices** | Yes | Limited | Yes |

**Table 1: Comparing Operational and Institutional Factors Related to 3S**

As a concrete example of potential synergies and their limits at the operational level, it is possible to look at the technical systems needed for safeguards. (See Table 2.) Containment and surveillance (C/S) is essential for safeguards and security, but it is not central for safety, and it can be a problem to the extent it may impede access. However, radiation monitors for radionuclide releases in the facility as well as other surveillance activities are of interest and are also relevant for safety.

Nondestructive and destructive assay (NDA and DA) are useful for safeguards and security, but are generally of limited value for safety purposes. However, in looking more broadly at environmental health and safety there are some commonalities. In some cases, it can be beneficial from a regulatory perspective to better characterize waste streams including, for example, spent fuel through NDA techniques. Such efforts can improve waste management and licensing efforts as well as maximize repository capacity. Similarly, environmental sampling (ES) has clear safeguards relevance and is also needed for safety. It may play a role in security as well.

The same holds for process monitoring. Generally speaking, the operator has a strong interest in process monitoring and control and will seek to maximize capabilities within the available budget. This

type of information can support both security and safeguards. However, for international safeguards, protection of intellectual property is a significant concern in this context, and could well limit the use of this data for 3S purposes.

For both safety and security purposes, design verification will be important. A detailed understanding of plant design and operation is of fundamental importance to safety. Maintaining continuity of understanding of changes over the lifetime of a facility will be important to ensure safeguards systems remain effective and the basis of safety assessments remain valid. Plant security systems will also need to take account of penetrations and relevant modifications that might introduce vulnerabilities.

| Safeguards Measures/Capacities | Safety | Security |
|---|---|---|
| **C/S** | Yes | Yes |
| **DA** | Yes? | Yes |
| **NDA** | Yes? | Yes |
| **ES** | Yes? | Maybe |
| **Process Monitoring** | Yes | Yes |
| **Design Verification** | Yes | Yes |

**Table 2: Safeguards Measures/Capacities and their Relevance to Safety and Security**

### A 3S Technology R&D Program for Improving Nuclear Governance

Addressing the challenges of enhanced safety, security and safeguards in the post-Fukushima environment will require good nuclear governance. Statesmanship at the United Nations, the Agency and among key nuclear States will be necessary to ensure that these objectives are balanced, that the capabilities of the Agency to carry out its safeguards mission are improved and that decisions on expanding roles and missions are wisely deliberated and taken. However, if the Agency and the international community are to be able to address the challenges they confront now and in the future, there are technical and issues that will have to be addressed.

Improving the IAEA's capabilities to meet current safeguard requirements will continue to be necessary because of the need to enhance efficiencies; to better address current requirements that are not optimized; and to be positioned to deal with future proliferation challenges. Meeting these challenges—current and future, anticipated and unanticipated—will require a defense-in-depth approach that includes:

- state-of-the-art instrumentation and methodologies for materials measurement, accounting and tracking, including sensor platform integration;
- enhanced containment and surveillance, including portal and area radiation monitoring, and measures to assure the absence of materials or radiation signals;
- integration of access denial and transparency elements of physical protection and safeguards; and
- integration of traditional process monitoring with non-traditional indicators, such as detection of radiation signals where they should not be, questionable movement of equipment and people, etc.

To support such an approach, it will be necessary to build upon the Next Generation Safeguards Initiative and other programs to create a flexible and adaptive technology base for advanced safeguards; to develop a new generation of inspectors and analysts; and to modernize the Agency's safeguards infrastructure. A technology R&D program should investigate next generation technologies to determine the best near term detection, measurement and forensic technologies, along with analytic tools and methods for development. It should also explore issues relevant to safety and security including:

- integrated facility design to enable advanced safeguards, physical security and safety measures, (i.e., safeguards-, security- and safety- by-design) and to optimize proliferation resistance throughout operational life cycles;

- intrinsic transparency in facility operations in support of 3S objectives;
- more robust integration of physical protection and safeguards, and an effort to bring safety into the equation; and
- systems analysis to evaluate design tradeoffs between facility operations, safety, security and safeguards effectiveness and cost, as well as to assess the effectiveness of an integrated system as a whole.

These areas could become a focus for the Agency's technical cooperation, which would strengthen implementation.

**Conclusions**

However attractive these concepts, the feasibility, costs, operational impacts and other possible effects of such approaches have not yet been demonstrated and will need to be analyzed further. If the issues identified above, and any others that may come to the fore, can be addressed in design in an optimal fashion without undue construction and operational costs, they can be successful. They could in principle strengthen all three objectives—safeguards, security and safety—by building on common systems and protocols, while minimizing tensions, for example, between safety and security. However, they may not be successful. In addition to further analysis, there is a need to demonstrate a "business case" for safeguards-, security- and safety-by-design.

**Hybrid K-Edge Densitometry Research at Oak Ridge National Laboratory**

Robert D. McElroy, Jr, Stephen Croft, Catherine E. Romano, and Tyler Guzzardo

*Safeguards & Security Technology, Global Nuclear Security Technology Division*
*Oak Ridge National Laboratory, Oak ridge, Tennessee, USA*

Hybrid x-ray fluorescence and K-edge densitometry (HKED) is used at nuclear fuel reprocessing facilities to determine the concentrations of uranium and plutonium in in the input accountability vessel. The uranium concentration is determined by K-edge densitometry, that is, from the step-difference in transmission on either side of the K-absorption edge measured using a continuous x-ray spectrum on a vial of solution defining a well-known geometry. The relative plutonium-to-uranium concentration is obtained from the relative strength of the x-ray-induced K-shell x-ray fluorescence production measured in energy-dispersive mode.

As commonly applied at current commercial reprocessing facilities, the plutonium-to-uranium ratio is about 1%, and the uranium concentration is typically in the range 100 to 300 g/L. Under these conditions HKED provides highly accurate results (with uncertainties of 0.2% on uranium and 0.7% on plutonium) more rapidly (combined assay time of about 1 h) and more economically than chemical methods and without secondary radioactive waste creation. The HKED technique was conceived and pioneered at KfK Karlsruhe by Herbert Ottmar and his coworkers. It has proved to be extremely successful blend of an almost absolute measurement technique and a robust relative method.

Extreme accuracy is required for the calibration requiring application of a number of small but important correction factors. Understanding the behavior of the calibration solutions and the nature and interplay of the correction factors for real process solutions remains the focus of research. Very few existing centers have the requisite multidisciplinary expertise base and facilities to maintain and develop the HKED method. In addition to the need to support inspection agencies and vendors with respect to the current application space, there is also a pressing need for a vibrant capability to support emerging needs and create and test alternative procedures and algorithms. For instance, advanced nuclear fuels enable far higher burn-ups today than in the past, and the expectation is that this trend will continue, creating challenges to the original and proven implementation approach of the HKED. The introduction of mixed oxide fuels is another shift in application space and draws attention to greater fission product activity and the quantification of minor actinides. The role of HKED in the front-end accountancy at pyroprocessing facilities is also under investigation.

In a move to address these important safeguards issues, Oak Ridge National Laboratory has installed an HKED at the Radiochemical Engineering Development Center. In this paper the HKED instrument and initial calibration measurements are described along with plans for future research and the establishment of an international user facility for research and hands-on training.

Keywords: Densitometry, XRF, HKED

# Simulations and Preparation for PNAR Measurements of Fugen Fuel

**J.M. Eigenbrodt[1,2], W.S. Charlton[2], H.O. Menlove[1], S.J. Tobin[1], M. Seya[3], H.R. Trellue[1], and A.M. Bolind[3]**

[1] Los Alamos National Laboratory
Los Alamos, New Mexico, U.S.
E-mail: jeigenbrodt@lanl.gov

[2] Texas A&M University
[3] Japan Atomic Energy Agency

***Abstract:***

*The Next Generation Safeguards Initiative Spent Fuel (NGSI-SF) Project1 will be deploying a non-destructive assay (NDA) system to measure heavy water moderated, light water cooled Fugen reactor fuel assemblies. The NDA system combines the following NDA techniques: (1) Passive Neutron Albedo Reactivity, (2) total neutron, (3) total gamma and (4) three fission chambers based neutron detectors sensitive to different parts of the neutron energy spectrum. Before deploying this system to Fugen, a variety of fresh fuel assembly comprised of differing numbers of low enriched rods (3.2%, 0.7%, 0.3% and 3.2% with Gd) will be measured at Los Alamos National Laboratory in water-based experiments that will used a Cf source place inside the assembly to provide a driving neutron source. The sensitivity of the neutron portion of the system will be assessed to positioning of the assembly inside the detector as well as to the removal/replacement of pins. Furthermore the variation in the relative intensity of the neutron flux in the thermal, epithermal and fast regions will be benchmarked. Deployment of this detector system at the Fugen reactor site is expected in the summer of 2013.*

**Keywords:** spent fuel; fission chambers; PNAR; NDA measurements

## 1. PNAR Detector

A Passive Neutron Albedo Resonance (PNAR) detector measures the multiplication of radioactive nuclear materials by comparing two measurements of the same material utilizing the intrinsic neutron emission to self-interrogate the nuclear material. One measurement is taken in a high multiplying region (in this case, with no cadmium liner), and the other in a low multiplying region (in this case, with a cadmium liner between the fuel and the fission chamber detectors). The ratio of these two measurements is called the Cd ratio, and has been shown to scale with fissile content[2,3].

**The PNAR detector as built for measurements of Fugen fuel consists of three ion chambers and twelve fission chambers, with the detector divided into three regions. The design of the PNAR detector is shown below in**

Figure 1.

Regions 1 and 2 are part of the high neutron multiplying region. These regions are filled with polyethylene and have no cadmium between the fuel and the detectors. Region 1 has three fission chambers, and region 2 has three ion chambers. The ion chambers are used to measure the total gamma emission from which the burnup along the axial length of the fuel can be inferred, and possibly for determining information about the position of the assembly within the detector. Region 3 is the low neutron multiplying region. In contrast to region 1, region 3 is mostly air and has a cadmium liner between the fuel and the fission chambers. Region 3 has three sets each of three fission chambers. These fission chambers are classified as fast flux monitors (FFM), cadmium covered fission chambers and bare fission chambers. Each of these sets of fission chambers has been tailored to measure different sections of the neutron energy spectrum. PNAR uses the ratio of fission chambers

in the high multiplying section to those in the low multiplying section to measure the multiplication of the assembly. In this paper, the PNAR ratio is calculated by dividing the sum of the three fission chambers in region 1 by the sum of the three fast flux monitors. The fission chambers in Region 3 will also be used to analyse different parts of the neutron energy spectrum.

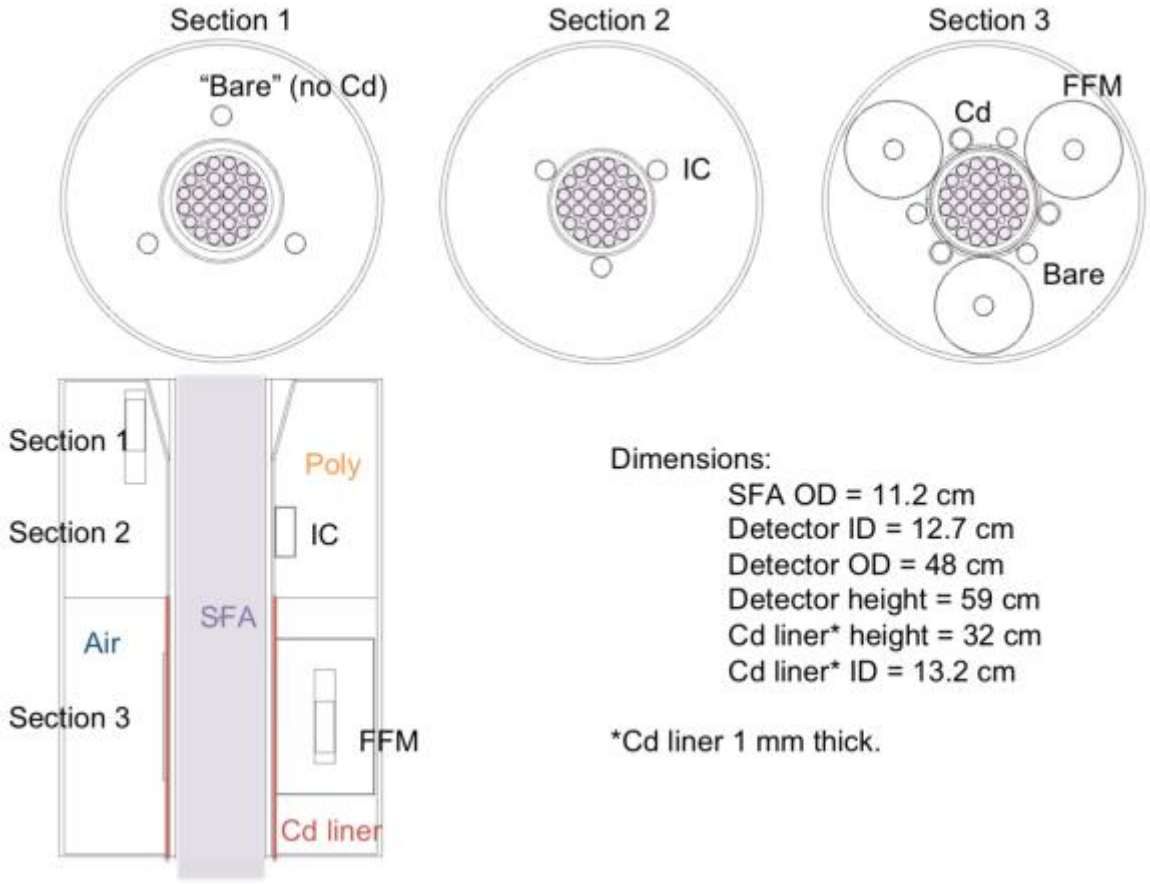


**Figure 1:** PNAR detector system as built for use at the Fugen site in Japan. SFA is the spent fuel assembly, IC are ion chambers, and OD and ID are outer and inner diameter, respectively.

## 2. Fugen Measurements

Measurements are scheduled to be performed with the PNAR detector on up to 9 different assemblies at Fugen site in Tsuruga, Fukui, Japan June 17-28, 2013.

### 2.1. Fuel Info

The Fugen Advanced Thermal Reactor in Tsuruga, Fukui, Japan was built to test the effects of using MOX fuel in a heavy-water moderated reactor. It operated from 1978 to 2003. It is a heavy-water moderated, light-water cooled reactor which was loaded with varying mixtures of low-enriched uranium (LEU) and mixed-oxide (MOX) assemblies[4, 5].

#### 2.1.1. Fugen Fuel Geometry

The Fugen reactor holds 224 assemblies, each of which has an effective fuel length of 3.7 cm, a radius of 5.6 cm, and contains 28 fuel rods. A view of a standard Fugen fuel assembly is shown below in Figure 2.

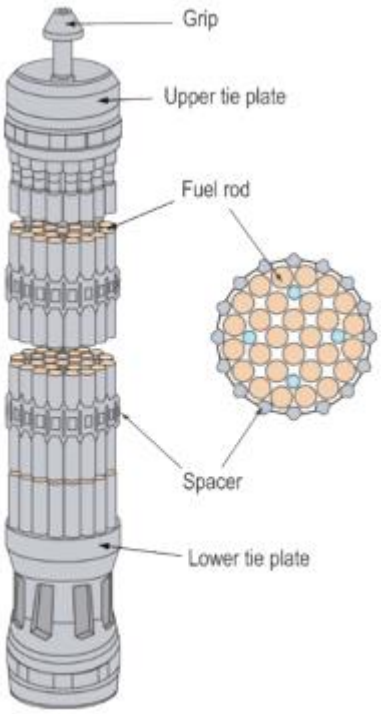


**Figure 2:** Schematic view of a standard Fugen fuel assembly[4]

### 2.1.2. Fugen Fuel Assemblies Available

The Japan Atomic Energy Agency (JAEA) and the Fugen site have provided nine assemblies to be measured with the PNAR detector. These assemblies include both LEU and MOX fuel, and include a wide range of burnups and cooling times based on the fuel still present at the Fugen Site. Basic information for these assemblies is shown below in Table 1.

**Table 1:** Fuel Available for Measurement at Fugen Site

| Assembly Name | Fuel Type | Burnup (MWd/MT) | Cooling Time (yrs) |
|---|---|---|---|
| M1 | MOX | 3,682 | 10 |
| M2 | MOX | 7,078 | 10 |
| M3 | MOX | 12,339 | 10 |
| M4 | MOX | 15,224 | 10 |
| M5 | MOX | 15,167 | 15 |
| M6 | MOX | 15,210 | 20 |
| M7 | MOX | 19,159 | 20 |
| L1 | LEU | 1,042 | 10 |
| L2 | LEU | 6,122 | 10 |

## 2.2. Measurement Simulations

The burnup of several of the assemblies listed in Table 1 were simulated using MonteBurns[5]. Subsequently, the anticipated response of the PNAR detector was simulated with MCNPX[6]. Fuel assemblies were selected for simulations based on information available on each fuel assembly. The Monteburns fuel simulations were performed with fuel burned to match burnup, cooling time, and a limited amount of destructive analysis information. JAEA and Fugen have recently provided more detailed information about fuel loading and power history for all assemblies listed in Table 1. More detailed fuel simulations will be performed in the future with the supplemental information recently

provided. The source term for each simulation was calculated using the concentrations of $^{244}$Cm and $^{240}$Pu in each assembly. The count rates in all tubes and PNAR ratios are shown in Table 2.

**Table 2:** Fugen Fuel Simulation Results

| Assembly Name | L2 | M1 | M2 | M4 | M6 | M7 |
|---|---|---|---|---|---|---|
| Burnup (GWd/MT) | 7 | 3 | 7 | 15 | 15 | 19 |
| Cooling Time (years) | 10 | 10 | 10 | 10 | 20 | 20 |
| Section 1 (cts/s) | 2.71 | 33.8 | 382 | 1818 | 762 | 547 |
| FFM (cts/s) | 0.187 | 2.34 | 26.5 | 129 | 55.5 | 41.1 |
| Bare (cts/s) | 0.242 | 2.96 | 33.9 | 164 | 71.0 | 52.8 |
| Cd (cts/s) | 0.018 | 0.228 | 2.53 | 12.3 | 5.34 | 3.80 |
| PNAR | 14.50 | 14.45 | 14.43 | 14.12 | 13.72 | 13.29 |

The multiplication of each Fugen assembly is compared to its simulated PNAR ratio below in Figure 3.

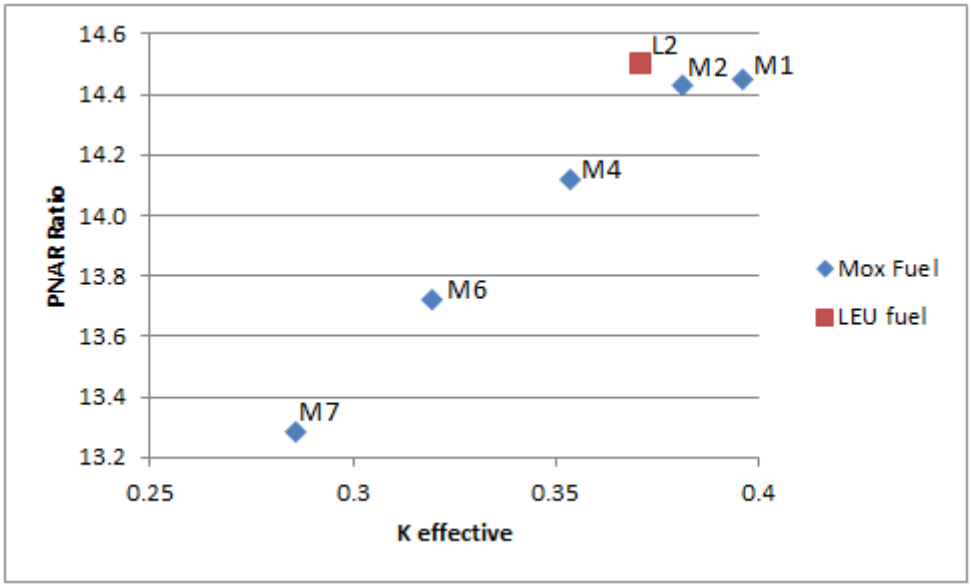


**Figure 3:** Multiplication and PNAR ratios for each simulated Fugen assembly.

## 3. Los Alamos National Laboratory Measurements

Additional measurements are to be performed at Los Alamos National Laboratory (LANL) with the PNAR detector on fresh pressurized-water reactor (PWR) fuel pins. The measurements at LANL will be made with fuel of well-known composition. Through benchmarking these measurements we gain experience with our capability to simulate PNAR with MCNPX. Furthermore, the LANL setup will provide an additional set of fuel enrichments and assembly designs.

### 3.1. Fuel Information

The PWR fresh fuel pins available for these measurements are shown in Table 3. They include LEU pins, depleted uranium (DU) pins, and LEU pins with gadolinium to simulate fuel poisons such as fission products.

**Table 3:** LANL Fresh Fuel Pins

| | LEU | DU | Gd rod |
|---|---|---|---|
| Number of rods | 204 | 204 | 12 |
| Pellet density (g/cm3) | 10.48 | 10.48 | 10.48 |
| Pellet O.D. (cm) | 0.851 | 0.851 | 0.877 |
| Rod O.D. (cm) | 1.08 | 1.08 | 1.08 |
| Cladding Material | Zr-2 | Zr-2 | Zr-4 |
| Cladding thickness (cm) | 0.115 | 0.115 | 0.064 |
| Rod active length (cm) | 103.5 | 103.5 | 104 |
| U-235 enrichment (wt %) | 3.19 | 0.2 | 3.28 |
| Gd wt% | 0 | 0 | 5.12 |

## 3.2. Assembly Designs

As stated above, the PNAR signal, the Cd ratio, is a ratio of the count rates measured in both high and low multiplying regions. An NDA instrument designer cannot very easily increase the multiplication in the high multiplying region since the spent fuel assembly in water is high multiplying. Hence, reducing the multiplication in the low multiplying section is a key design focus. Cd is excellent at absorbing thermal neutrons so it is an excellent material selection to reduce the multiplication. The NDA instrument designer would like to position the Cd as close as possible to the fuel since a water layer around the assembly reduces the ability of the Cd layer to change the multiplication in the assembly. How close the Cd can be positioned in reality depends on the regulatory decisions of the relevant authorities. In the case of the Fugen fuel, a water gap of 15 mm (7.5 mm on each side) was used in the fuel storage racks, and as such, this same water gap was used in the PNAR design. One fuel rack used at LANL matched the outer diameter of the Fugen fuel. A second rack was fabricated at LANL that was somewhat larger. This was done to reduce the water gap to that which is more typical for commercial PWR and BWR fuel storage racks. This second gap had a water gap of 6.4 mm (3.2 mm on each side).

### 3.2.1 Assembly 1 – Fugen Fuel

The first assembly was designed to represent the Fugen fuel, but with PWR pins. In order to keep the same mass of fuel per length of the assembly, 60 PWR pins were used in this design. The assembly radius was kept the same as the Fugen fuel assemblies – 5.6 cm. This fuel assembly design is shown in Figure 4. In Figure 4 and Figure 5, the rectangles in the outer ring of fuel pins are for the spacer screws to be used to set the location of the assembly within the detector.

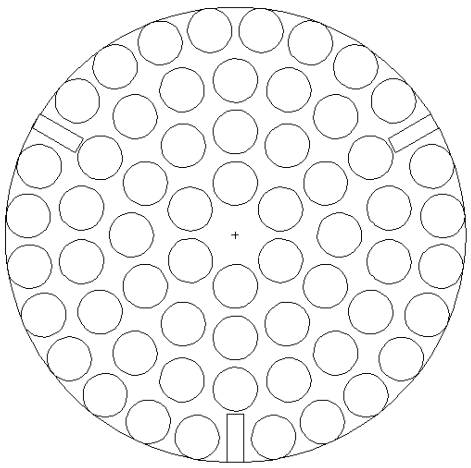

**Figure 4:** Assembly design 1 (representing Fugen fuel).

### 3.2.2 Assembly 2 – PWR Fuel

The second assembly was designed to represent more typical PWR fuel, but within the constraints provided by the size and shape of the as-built PNAR detector. This design was chosen to have a water gap between the fuel of 1/8" – similar to the spacing found in PWR spent fuel storage racks. This resulted in an assembly radius of 6 cm. The assembly was also chosen to have a similar fuel to moderator ratio as typical PWR fuel, which resulted in having 51 PWR pins. This fuel assembly design is shown below in Figure 5.

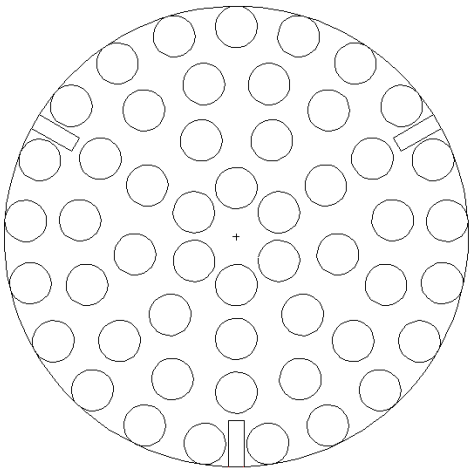

**Figure 5:** Assembly design 2 (representing PWR fuel).

## 3.3. Measurement Simulations

MCNPX Simulations were performed to inform possible measurement setups to be used at LANL. For both assembly designs, sets of simulations were performed to investigate measurement changes as a result of changes in effective enrichment, position within the detector, and gadolinium concentration in the fuel.

### 3.3.1. Measurement Setup

PNAR measurements at LANL using PWR fresh fuel pins will be performed in a tank of water. The PNAR detector will be supported on a stand to allow the fuel to extend both above and below the detector. There will be two grid plates for each assembly to maintain pin spacing, and they will be connected with three tie rods. A $^{252}$Cf source with source strength of $5.0 \times 10^6$ n/s (on May 1, 2013) will be used to drive the measurements. It will be centered between PNAR sections 1 and 3. The grid plates and source tube are shown in Figure 6.

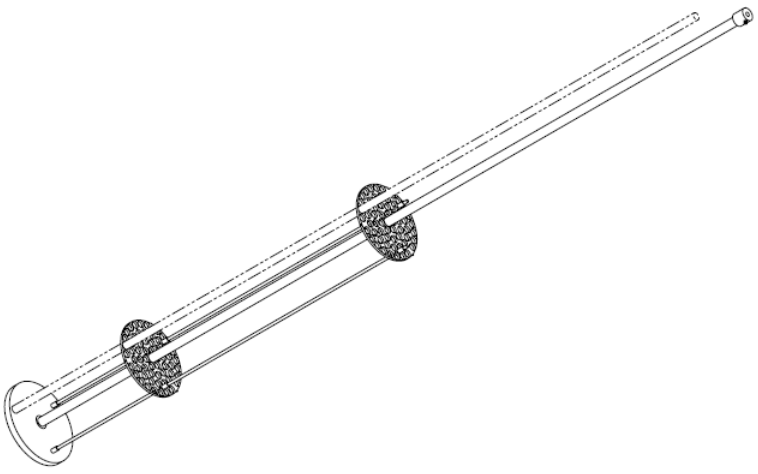

**Figure 6:** Assembly grid plates, tie rods (one shown dashed), and source tube (center)[7].

### 3.3.2. Effective Enrichment Simulations

Five different effective enrichment setups were simulated for each assembly design by evenly mixing depleted uranium and low-enriched uranium fuel pins throughout the assembly. The pin arrangements for assemblies 1 and 2 are shown below in Figure 7 and Figure 8, respectively.

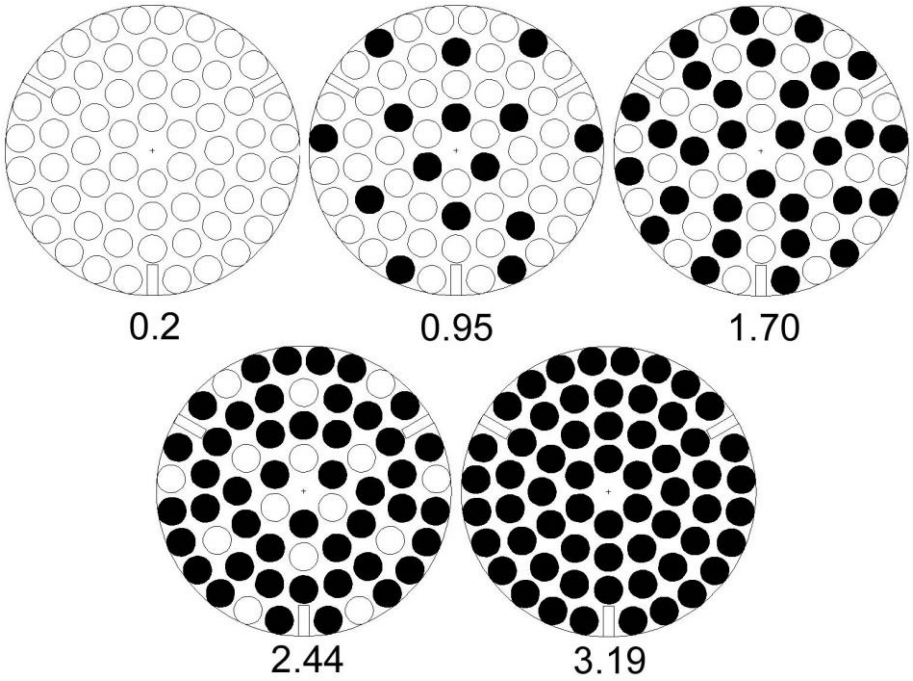


**Figure 7:** Enrichment arrangements for assembly 1, where white are DU pins and black are LEU pins.

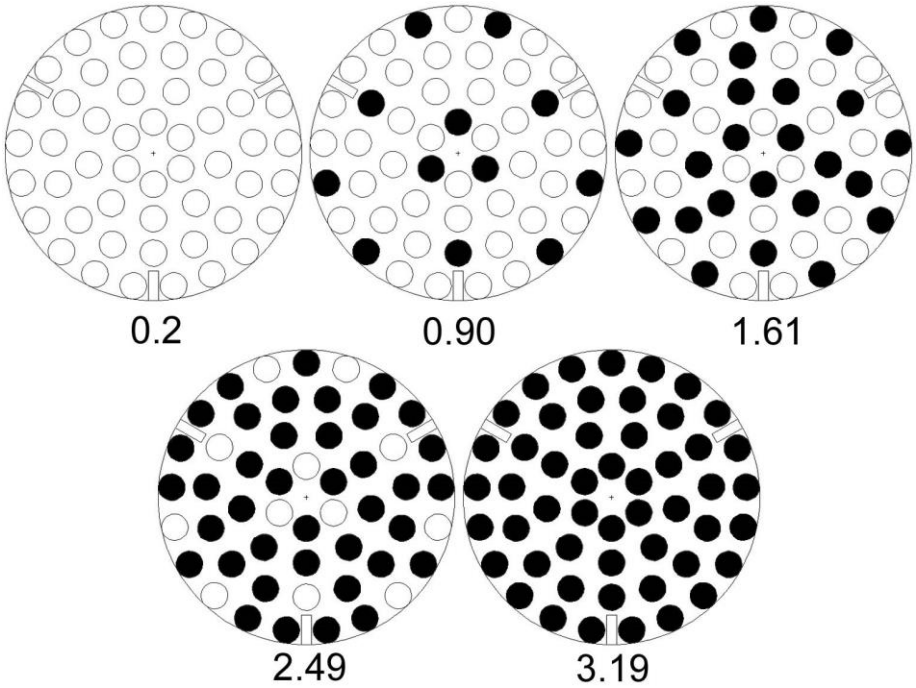


**Figure 8:** Enrichment arrangements for assembly 2 where white are DU pins and black are LEU pins.

PNAR ratio measurements for both assemblies between 0.2 and 3.19 wt% $^{235}$U are shown below in Figure 9. Counting statistics for 1.5 hour measurements are below 0.3% for the PNAR Cd ratio.

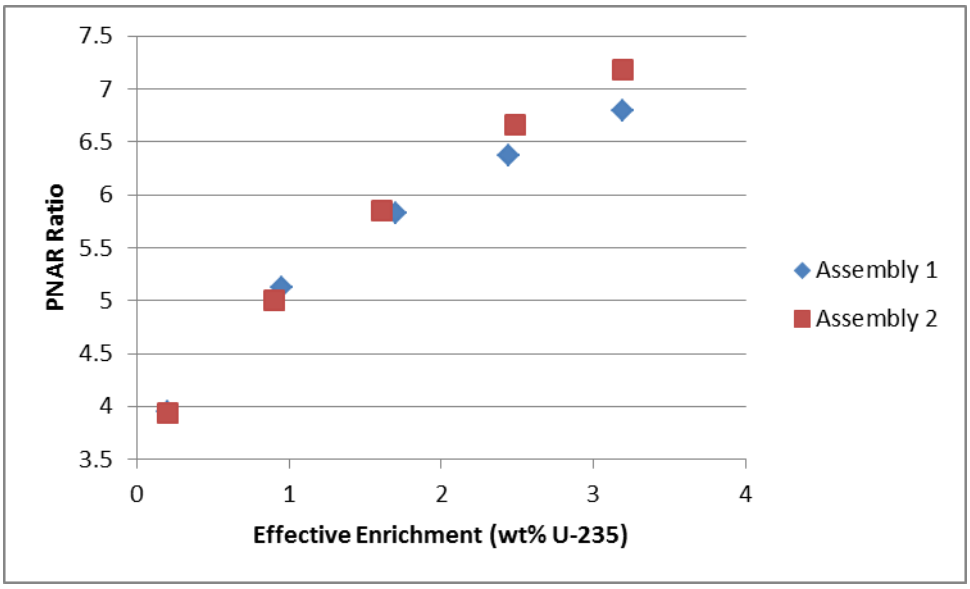


**Figure 9:** PNAR ratios vs. effective enrichment for both assembly designs.

### 3.3.3. Future Simulations

Simulations are also being performed to investigate the effects of fuel assembly position within the detector and fuel poisons.

The largest expected systematic uncertainty in the Fugen measurements is the position of the assembly inside the detector. This will be investigated during the LANL measurements by using the three screws in each grid plate to precisely set and determine the position of the assembly within the detector during measurements. Six measurements will be performed on each assembly design at different locations within the detector, as shown in Figure 10. There assembly will be displaced between the center of the detector and the edge of the detector along three vectors: 0°, 30°, and 60° with respect to one fast flux monitor in region 3. Since the detector design is symmetrical in increments of 60°, this will cover displacements in 30° increments around the entire detector. Assembly 1 has a gap of 0.7 cm between the fuel and the detector, and will be displaced by 0.35 and 0.7 cm. Assembly 2 has a gap of 0.3 cm between the fuel and the detector, and will be displaced only by 0.3 cm along each vector. In Figure 10 below, positions 2, 4, and 6 are with the assembly directly against the detector, and will be measured for both assemblies. Positions 1, 3, and 5 are half-way between the center of the detector and the edge, and are only to be measured for assembly 1.

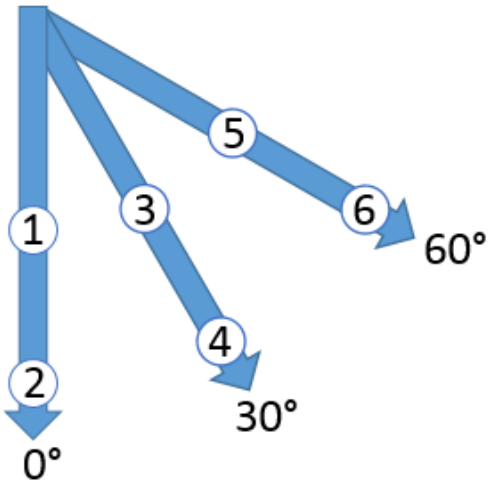


**Figure 10:** Positions to be measured at LANL.

The gadolinium-laced fuel pins will be used to investigate the effect of fuel poisons, such as fission products, on PNAR measurements. Two gadolinium measurements will be performed with each assembly design: with six and twelve gadolinium-laced fuel pins spread throughout the assembly with the rest of the assembly filled with LEU pins.

## 4. Future Work

There are several areas where work is still on-going. Measurements will be performed at both LANL and Fugen site by the end of June. More work needs to be done on further detailed simulations of Fugen fuel given the additional information the JAEA has provided. Additional simulations may be performed for LANL fresh fuel measurements in other configurations to test sensitivity to aspects such as radial dependence on enrichment. Further analysis will also be performed on the signatures available with the bare and cadmium covered fission chambers to obtain information about different parts of the neutron energy spectrum.

## 5. ACKNOWLEDGEMENTS

The authors would like to acknowledge the support of Fugen Decommissioning Engineering Center of JAEA, Japan Safeguards Office of MEXT (Japanese Government), Next Generation Safeguards Initiative (NGSI), Office of Nonproliferation and International Security (NIS), National Nuclear Security Administration (NNSA).

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# ORNL SAFEGUARDS LABORATORY: A CENTER OF EXCELLENCE IN NONDESTRUCTIVE ASSAY AND A TRAINING CENTER FOR INTERNATIONAL SAFEGUARDS AND SECURITY

Angela Lousteau, Stephen Croft, Robert McElroy, Chris Pickett
Oak Ridge National Laboratory
Nuclear Security and Isotope Technology Division
Safeguards and Security Technology
Safeguards Laboratory
May 2013

**Abstract**

The Oak Ridge National Laboratory (ORNL) Safeguards Laboratory has recently expanded its capabilities to include a unique collection of large sophisticated Non-Destructive Assay (NDA) instruments that can be utilized as a flexible scientific asset to support international nuclear safeguards. The expansion of the Safeguards Laboratory includes new laboratory space with the addition of approximately $6M of NDA systems including an installed Hybrid K-Edge Densitometer (HKED) (of the type currently used by the IAEA in reprocessing facilities), a $^{252}$Cf Shuffler (used for assaying waste drums and other items), a Segmented Gamma Scanner (SGS), and a large-volume differential die-away system (DDA) (for finding SNM in shielded packages and quantification of fissile content). These instruments are of the types typically installed at operating nuclear facilities worldwide; however, access to them for research and training is often limited because of operational and security constraints. The addition of these new instruments to the Safeguards Laboratory enables ORNL to provide an accessible user facility to the international community for R&D, training, and independent testing in support of international nuclear safeguards. For instance, opportunities exist for the independent evaluation and development of advanced assay algorithms and concepts, to promote best practices, to mimic and mitigate field problems, to improve and study modeling tools and techniques, to support industry, and to maintain and enhance subject matter expertise essential in being able to advise governments from a position of authentic authority, and to actively participate with international organizations in the maintenance and creation of consensus standards and guides. This paper discusses the safeguards-related goals and objectives associated with the expansion of the Oak Ridge Safeguards Laboratory and the proposed training and R&D work associated with the new equipment.

**Background**

The ORNL Safeguards Laboratory is operated by the Safeguards and Security Technology (SST) Group in the Nuclear Security and Isotope Technology Division (NSITD). This facility supports R&D and training for all elements associated with domestic and international safeguards and global nuclear security. Over 2000 people have been trained in a variety of specialisms during the past four years and several collaborative research and development (R&D) projects have been initiated with university and national laboratory partners. Much of the current training encompasses non-destructive assay (NDA) measurement capabilities for the control and accountancy of nuclear material. The laboratory provides support including equipment, nuclear materials and other radioactive sources, and expertise to nuclear nonproliferation programs to improve effectiveness and efficiency of international safeguards regimes.


**Fig.1. Off-site NDA Training**

**Fig.2. Hands-on Training in Lab**

The Safeguards Laboratory provides a platform and integration center for coordinating radiological/nuclear detection and measurement activities and training. The Safeguards Laboratory provides the tools needed for these training modules and includes controlled access to nuclear materials and calibration standards for both U.S. domestic and foreign visitors. There are a number of standard radiation detectors (NaI, CZT, HPGe, HM-5, AWCC), multi-channel analyzers (MCA-166, DART, DigiDART, Inspector 2000, etc), and software packages (MGAU, NaIGEM, U/Pu, HMS4, ISOCS, WinU235, WinUF6, INCC, Genie 2000, Maestro, etc.) that are available as training modules or for use in the Safeguards Laboratory. These instruments and software packages are widely used by inspection agencies around the globe and also serve to train users on general detection principles.

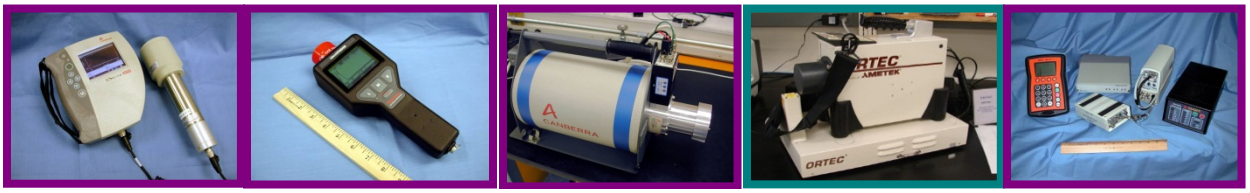


**Fig.3. Examples of Portable Instruments for Use or Training**

Controlled foreign national access to special nuclear materials and other radiation sources provides international users the ability perform hands-on activities using real materials. The Safeguards Laboratory currently maintains a CRM-969 set of uranium oxide standards consisting of 5 oxide cans of varying enrichment (depleted through 4.46%), a CRM-146 set of uranium oxide standards consisting of 3 oxide cans (20.11%, 52.49%, and 93.17% $^{235}$U), and many other secondary standards (in point, line, and area geometries) typically used for the calibration of holdup equipment and to perform learning exercises. In addition, ORNL has plans to fabricate sharable standards of ultra-pure $^{233}$U and is hoping to develop a new set of slightly enriched uranium SEU) standards (5% - 19.7%) similar in form to the SRM-969 and SRM-146 suitcase sets. The availability of these materials (especially for international partners) is quite unique. They are also available for commercial vendor testing.

**Fig.4. Selection of Accountable Nuclear Material**

**Fig.5. Variety of Calibration and Check Sources**

Traditionally, the Safeguards Laboratory has conducted courses on a number of safeguards topics such as “Fundamentals of Radiation Detection”, which covers the basic physics behind radiation measurements and the basic equipment/hardware used in radiation detection; “Handheld Equipment Used for Radiation Measurements”, which covers the theory and basic use of a number of handheld instruments used by operators and inspectors (HM-5, U/Pu, Inspector 1000, Interceptor, and more); “Holdup Measurements: A Hands-on Approach”, which provides an in-depth class on the use of HMS4 for holdup measurements; “Overview of Uranium Enrichment Measurement Techniques”, which traditionally covers the use of WinU235, NaIGEM, MGAU (U/Pu), and PC-FRAM for enrichment measurements; and “NDA Applications for International Safeguards”, which includes an overview of the above mentioned courses and typically covers the Active Well Coincidence Counter (AWCC) and In-Situ Object Counting System (ISOCS). Occasionally, the Safeguards Laboratory is asked to conduct courses ad-hoc; and therefore is also capable of customizing courses for potential or perspective users such as security personnel or emergency responders. Serving the U.S. international outreach and engagement program is a prominent role of the laboratory.

**Fig.6. An NDA Course in Progress**

**Fig.7. Uranium Holdup Practical Exercise**

In addition to these classes, the Safeguards Laboratory has developed many customized courses for customers including topics such as standards development, advanced AWCC, and Segmented Gamma Scanner (SGS). Recently, the Safeguards Laboratory has expanded to include not only additional safeguards topics, but developed and/or hosted training in multiple other nonproliferation topical areas such as inspection fundamentals, containment and surveillance, security systems, and radiation inspection systems. A suite of equipment for these activities has also been accumulated.

**Fig.8. Large-Volume Active Well Coincidence Counter**

**Fig.9. In-Situ Object Counting System**

As technology advances and customer needs evolve, the Safeguards Laboratory continues to grow and improve by providing new facilities for evaluation, testing, and calibration of new safeguards technologies, develop new safeguards testing and validation procedures, and update and add new training capabilities and technical support to local, national, and international

organization. To support this increasing need, the Safeguards Laboratory has begun offering new courses and is physically expanding detection capabilities as described below.

## Expanded Training Topics

*Inspection Exercise Training*

The Safeguards Laboratory and the neighboring Containment and Surveillance (C/S) Laboratory work together to provide an introduction to inspection activities as part of a workshop focused on international inspections. The full day (all hands-on) course covered topics such as physical inventory taking, complimentary access, design information verification (DIV), seals and tags, enrichment verifications, and other quantitative measurements (HMS4 and AWCC).


*Containment and Surveillance (C/S)*

The Safeguard Laboratory, together with the C/S Systems Laboratory (another asset of the STIC), also provides training on the use of seals and tags, remote and/or unattended monitoring, tracking, surveillance systems, and authentication of equipment and data for safeguarding nuclear materials. The training includes an overview of the purpose of containment and surveillance technology and how it can be utilized to establish a continuity of knowledge for assets of concern, as well as the use, application, and best practices related to the use of C/S tools. The C/S Laboratory is primarily used to evaluate and develop custom technology as well as integrate, mock up, and stage equipment for evaluation, for instance feed and withdrawal loops.

*Portal Monitor Fundamentals (PMF)*

The Second Line of Defense (SLD) is responsible for implementing and transitioning radiations portal monitors (RPMs) at border crossings, airports, and ports around the globe. The Radiation Detections Systems Team provides training on the fundamental basics regarding the instruments they deploy. The PMF course is offered a number of times a year and consists of a full two day course covering topics such as: basic principles of radiation and radiation detection, statistics, alarm rates, data analysis, and best practices. The students get a mix of lecture and hands-on time with an RPM.

## Expanded Facilities/ Potential Training Topics

The Safeguards Laboratory is currently expanding to include additional systems/equipment used to measure nuclear materials covering a larger portion of the nuclear fuel cycle. ORNL has recently installed the following NDA systems:

- Hybrid K-Edge Densitometer (HKED)
- Segmented Gamma Scanner (SGS)
- $^{252}$Cf Shuffler
- A differential die-away-based (DDA) package monitor

Within the U.S. this represents a unique opportunity to conduct R&D, support field work, and conduct training on such a range of key instrument types.

*Hybrid K-Edge Densitometer*

The HKED installed at the Radiological Engineering Development Center (REDC) is a system that combines K-edge densitometry with X-ray Fluorescence to determine the Pu/U ratio and concentration of samples taken from dissolver solutions in a process stream. With no sample preparation required, short turnaround times (~20-60 min), and equivalent measurement precision (sub %), the HKED used to provide near-real-time analysis as an alternative to traditional 'wet chemistry' with mass spectrometry measurements. The addition of the HKED provides a test center for upgrades and modifications to the HKED system, demonstrate the performance of new algorithms/ improvements, and validate new reference standards. This is particularly important since new material processing regimes and evolving fuel cycles have made it necessary to accommodate much higher levels of plutonium (relative to uranium) than in the past.

The Safeguards Laboratory also provides an opportunity for domestic and foreign partners to gain valuable experience and training operating the HKED. Currently, plans to develop a new training module, likely a remote training module, are underway. Trainees for the HKED will likely include international partners/facility operators/ inspectors, and university students from several nuclear engineering programs.

*Segmented Gamma Scanner*

For more than 40 years, SGSs have been crucial to safeguards and material control and accounting (MC&A) of uranium and plutonium. The SGS at ORNL is a gamma-ray based assay technique that provides quantitative mass values for U and Pu in 200 liter drums. The system provides accurate assay values through the use of a collimated high purity germanium detector. The container is examined as a number of vertical segments and the U and Pu masses

(activity is non SNM isotopes are of interest) are quantified in each vertical segment using various matrix correction techniques which account for the gamma-ray attenuation for each segment. The SGS utilizes a transmission source, typically $^{152}$Eu, to correct for attenuation effects within the item of interest, but can be used solely as a passive measurement system as well.

Current plans for the SGS include algorithm development, validating new measurement methodologies and standards, and exercising new modeling tools. Particular areas of interest include evaluation of numerical calibrations versus source-based calibrations, evaluation of end-effects, and optimization of the counting protocol (e.g., how many segments to measure, segment size, etc. since there has been a creep towards using fewer segments to gain higher sensitivity at the expense of accuracy).

The SGS will also provide a test bed to mimic and resolve in-field problems. In the near future, a tomographic gamma scanner will, we hope, be added to the SEL to further enhance capabilities fro quantitative mass assay of safeguards relevant items. The availability and access to an SGS provides another unique training capability for ORNL.

*Californium 252 Shuffler*

The $^{252}$Cf Shuffler is an active NDA measurement system that induces fission in a wide variety of samples containing special nuclear material through the rapid insertion/ removal of a strong $^{252}$Cf neutron source. After a few seconds of interrogation, the source is removed and delayed neutrons are counted. The irradiation/count cycle is repeated to provide the desired measurement precision. From an analysis of the observed delayed neutron count rates, the amount of fissionable nuclear material present can be determined.

The shuffler relies on fast neutrons which better penetrate large, dense, concentrations of uranium and plutonium than would a gamma-ray or thermal neutron interrogation technique. Small-item shufflers can provide highly accurate (<0.25%) mass results for kilogram quantities of highly enriched uranium oxides, metals, and alloy product materials.

Shufflers are also useful when the matrix of the material inside a drum or other container is of varying density, quantity, composition, and distribution (typically of waste assay) and can achieve detection levels on the order of 0.1 grams of $^{235}$U in a typical 20 minute assay. This versatility allows shuffler systems to be incorporated into a wide range of waste, safeguards, and MC&A programs.

A 200 liter drum shuffler has been installed in the SEL and is currently being brought into service. Current R&D plans for the shuffler include validating new SNM standards, improving U and Pu NDA measurement algorithms, and exploring the potential for $^{3}$He alternatives in

active interrogation systems. In that respect, the shuffler will serve as a development platform to evaluate new neutron detection technologies and methodologies. By making the $^{252}$Cf Shuffler available to users, ORNL will be providing an excellent advanced training opportunity for graduate students and professionals by providing additional research tools that can be used to study the validity and benefit of neutron imaging, for example.

*Package Monitor-*

The package monitor is a differential die-away (DDA) time measurement technique that utilizes a D-T generator and an array of $^{3}$He tubes to detect the presence of gamma-shielded SNM. The package monitor is used to identify the presence of SNM in packages where gamma shielding prevents traditional passive gamma measurement techniques. Additionally, the inclusion of the package monitor helps provide ORNL with another opportunity to explore R&D opportunities with our academic partners, including time-analysis methods and (n,γ) capabilities, and to develop new training modules on another active interrogation technique.

**Summary**

The instruments in the ORNL SEL are examples of large, complex devices which are often installed at nuclear facilities and therefore are difficult for trainees and R&D staff to access. These tools will provide ORNL SMEs the opportunity to independently test and develop advanced assay algorithms and concepts to promote best practices, support industry, and advise government and consensus standard organizations alike. A particular thrust will be the development and promotion through consensus working groups of realistic uncertainty qualification methods that are fit for purpose.

In summary, the Safeguards Laboratory at ORNL is growing to meet the needs of both domestic and foreign users. By employing the use of technology such as the HKED, the $^{252}$Cf Shuffler, the SGS, and the Package Monitor, the Safeguards Laboratory is preparing to meet the expanding challenges facing the global nuclear safeguards and security community. In combination with new training modules, the ORNL Safeguards Laboratory is expanding beyond its current leadership in passive NDA training and expertise to include active interrogation and other NDA techniques for the measurement and quantification of nuclear materials across the fuel cycle.

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# Field Use of In Situ Object Calibration Software (ISOCS) for Gamma Spectroscopy Based Nuclear Safeguards Applications

**Andrey Bosko, Gabriela Ilie, Ramkumar Venkataraman, William Russ, Frazier Bronson**

Canberra Industries Inc.
800 Research Parkway, Meriden, CT 06450, USA
E-mail: abosko@canberra.com

**Andriy Berlizov, Vladimir Nizhnik**

International Atomic Energy Agency
Vienna International Centre, PO Box 100, 1400 Vienna, Austria

***Abstract:***

*The ISOCS mathematical method developed by Canberra Industries has been widely used in various gamma spectrometry applications for computing High Purity Germanium (HPGe) detectors' efficiencies for a wide variety of source shapes and sizes. In a typical ISOCS application, the user inputs the source dimensions, the matrix composition and density, along with the source-to-detector distance, and then launches the efficiency computation. Recently, Canberra has developed a software utility for the International Atomic Energy Agency (IAEA) that simplifies, automates, and optimizes ISOCS based analysis in nuclear safeguards verification measurement campaigns. The new software utility called the Field ISOCS Utility is focused on the ease of use by IAEA inspectors in the field. The set up parameters for the measurement and analysis protocol are pre-defined by an expert in Headquarters. The inspector in the field then executes the analysis. The Setup Application in the software provides a tool for the expert to set up the analysis protocol for a given counting scenario. The inspector in the field runs the Analysis Application in the software which requires only minimal input. The Field ISOCS Utility can be used to execute the following analysis protocols: (i) running a simple ISOCS analysis for a given counting geometry and a measurement item, without changing the pre-defined input parameters; (ii) running ISOCS analyses by updating one or more variable parameters of the measurement item, e.g., the fill height of the sample, tare weight of the container etc.; and (iii) running a full blown efficiency optimization using the Advanced ISOCS Uncertainty Estimator (A-IUE) benchmarked by the data available in the measurement spectrum. The operation of the Field ISOCS software utility is described for the above mentioned measurement protocols. The functionalities of the Expert-Inspector features are explained in detail. Examples of measurement scenarios are presented.*

**Keywords:** ISOCS; safeguards; gamma spectrometry

## 1. Introduction

The ISOCS mathematical method developed by Canberra Industries [1] has been widely used in various gamma spectrometry applications for computing High Purity Germanium (HPGe) detectors' efficiencies for a wide variety of sample shapes and sizes. It has been routinely used by inspectors in the field for performing safeguards measurements. At that the efficiency calibration of the measurement geometry is a critical step in the achieving accurate quantification of special nuclear material (SNM) content. During the typical ISOCS calibration process the user is required to provide input data such as detector information, source to detector distance, dimensions of the item being measured, source matrix information, including the container fill height, material composition and density. This information is used to define the ISOCS geometry template, which is then used to calculate the full energy peak efficiency for a given counting geometry. In many cases the source and

geometry parameters are not available in advance and can be only obtained during the actual measurement in the field. Once the exact geometry data is obtained it needs to be used in the field to create a precise ISOCS calibration model. Some of the sample information may not even be available to the user, e.g., fill-height of material in closed container, material density etc. In this case, the user is left with the option of determining the best geometry parameters that are consistent with the measured data.

Considering that the safeguards measurement campaigns are performed under tight time constraints, any modification to the ISOCS efficiency calibration or manual geometry optimization by trial and error will introduce delays and limit the total number of measurements that can be performed during an inspection on site. A manual trial and error process will also introduce delays in obtaining the final analysis result because some of the measured data will need to be evaluated by the expert who is located off-site. In response to these difficulties, Canberra has recently developed a software utility for the International Atomic Energy Agency (IAEA) that simplifies, automates, and optimizes ISOCS based analysis in nuclear safeguards measurement campaigns. The new software utility called the Field ISOCS Utility is focused on the ease of use by IAEA inspectors in the field and is supposed to save time and efforts necessary to perform verification measurements.

The Field ISOCS Utility is intended to simplify the use of ISOCS calibration software in the field by introducing the Expert – Inspector concept to the measurement scenario. In this scenario an expert located in a central office and experienced in using ISOCS, will configure a set of geometry templates and analysis procedures, which then will be utilized by a user in the field. The Setup Application in the software provides a tool for the expert to set up the analysis protocol for a given counting scenario. The user in the field runs the Analysis Application in the software which requires only minimal input.

## 2. Field ISOCS Setup application

The Field ISOCS Setup application is intended for use by the Expert, who will use it to create a setup file containing all analysis parameters. This setup file determines the way the spectrum analysis is performed in the field by the Field ISOCS Analysis application. With the existing measurement approach the Expert typically prepares a number of generic ISOCS templates corresponding to the expected counting geometries and defines the analysis steps, which are then used by the inspector in the field when performing the measurement. Since the actual measurement geometry parameters represented by the generic ISOCS template may be varying from sample to sample, the Inspector often needs to recalculate and manually adjust some of the geometry parameters in the template prior to running the efficiency calibration. These manual steps inevitably lead to possible efficiency calibration errors and also increases the overall time needed to analyze the spectrum. The Field ISOCS Setup application simplifies the Inspectors' work and increases the efficiency of the measurement campaign.

There are several major pieces of information that can be stored in the Field ISOCS setup file. One of them is information about the analysis steps to be performed on the measured spectrum. These analysis steps typically include peak location, peak area calculation, background and efficiency correction, as well as nuclide identification and activity calculation algorithms. All analysis steps are combined in a single Analysis Sequence File (ASF), which is then used by Canberra's gamma-spectroscopy software Genie-2000 [2]. The ASF is typically prepared in advance by the Expert and then just used by the Inspector in the field after a gamma spectrum has been acquired.

The second piece of information which is required for successful spectrum analysis is the ISOCS geometry information. This geometry information is used to build the peak efficiency curve for a given counting geometry, and, therefore, needs to be defined quite precisely. A number of geometry templates is available in ISOCS, which can be used to create a reasonably accurate model for most of the counting geometries encountered in the field. One of such geometry templates representing a simple cylinder geometry is presented in Figure 1. Prior to performing efficiency calibration the user needs to define all dimensions, source-to-detector distance, materials, and densities, so that they closely represent the actual counting geometry. For example, the template shown in Figure 1 can be used to describe a nuclear material measured inside a cylindrical container, e.g., a drum, or a can. The geometry parameters are stored in a geometry file, which is used by ISOCS to run the efficiency calculations.

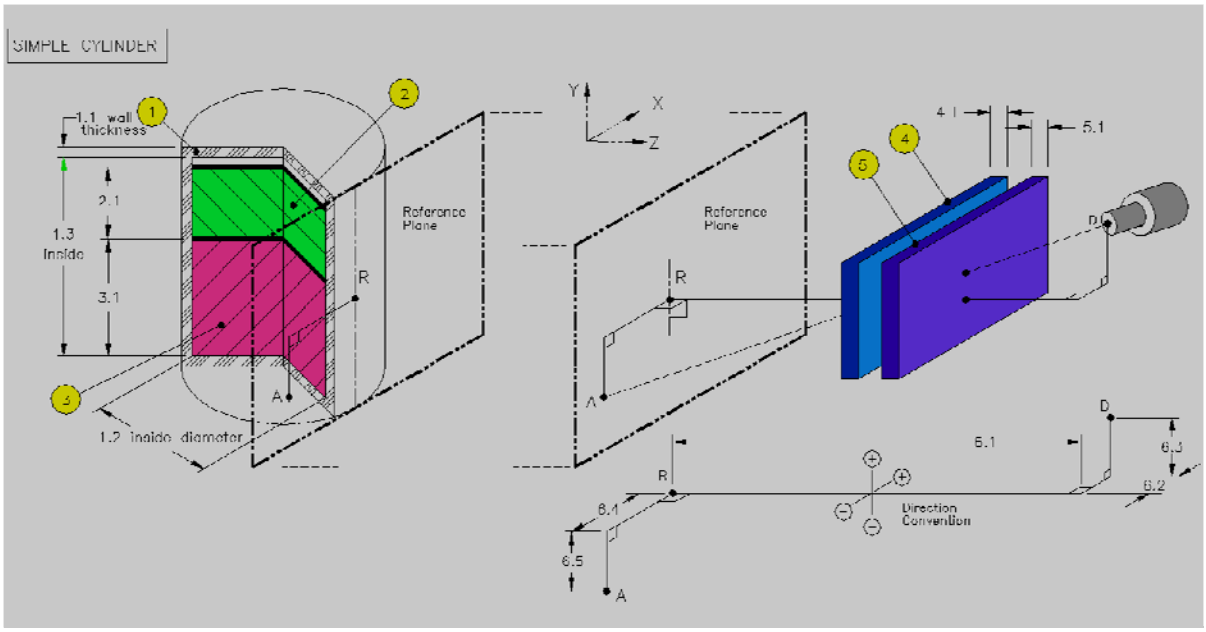


**Figure 1**: ISOCS template representing a simple cylinder geometry.

In general the Expert creates a number of generic ISOCS geometry files, which correspond to various measurement geometries that can be possibly encountered in the field. This is usually done during the planning phase of the measurement campaign taking into account nomenclature of typical geometries of nuclear material samples which should be verified. However, although general information about the expected counting geometry is usually known in advance, e.g., it may be known that a set of drums needs to be measured, the exact details, such as source-to-detector distance, density, fill height, material composition, container thickness, etc…, may not be available until the Inspector performs additional measurements or obtains additional information in the field. As a result, the geometry information which is available in a generic ISOCS geometry file created by the Expert will have to be manually updated in the field. The manual update steps start with gathering additional information, e.g., by using scales, ruler, thickness gauge, radiography information, site data provided by the facility operator, etc… Then this acquired data must be converted into geometry parameters of the ISOCS template file and manually entered using the ISOCS Geometry Composer. For example, gross weight, empty container weight, measured container dimensions and sample fill height can be used to calculate the material density. Often these calculations are performed separately on a piece of paper, or using one of the mathematical type applications (e.g., Microsoft® Excel) and then manually transferred into the ISOCS geometry file. Besides the fact that calculations must be done quite fast due to time constraints, the manual data transfer step by itself may lead to possible errors. Therefore, it was essential to give the Field ISOCS Setup application a capability of defining mathematical relationships between variable Sample parameters, which can only be obtained at the time of measurement, and various ISOCS template parameters, which need to be updated in a generic ISOCS geometry file based on the field data.

With the Field ISOCS Setup application the Expert can pre-define a list of Sample parameters that are obtained in the field. The Sample parameter can be anything that can be expressed as a numerical value, and which in turn is translated into a relevant ISOCS template parameter. Typical Sample parameters include various geometry dimensions that can be measured externally, weight, source-to-detector distance, elemental weight fractions in a material compound, uranium enrichment, etc… A set of formulae is then defined using a set of common arithmetic operators in the setup file, which converts the available Sample parameters values into the ISOCS template parameters values. Figure 2 displays such a conversion formula between one of the ISOCS template parameters (sample material density) and four different Sample parameters (gross weight, tare weight, container inner diameter, and matrix fill height).

The Field ISOCS Setup application can also be used to reference the ISOCS geometry optimization project file, which was defined by the Expert. If this is done, then during the spectrum analysis the

Advanced ISOCS Uncertainty Estimator (A-IUE) program [3] will be used to automatically optimize ISOCS geometry parameters using data obtained directly from the measured gamma-spectrum. In this case, the software will use spectral data, e.g., line activities for the isotopes present in the spectrum, or information obtained by some other means, for example, isotopic analysis in case of uranium or plutonium items, to determine the best set of ISOCS geometry parameters. These parameters will then be used to generate the peak efficiency curve for a given counting geometry and analyze the measured spectrum.

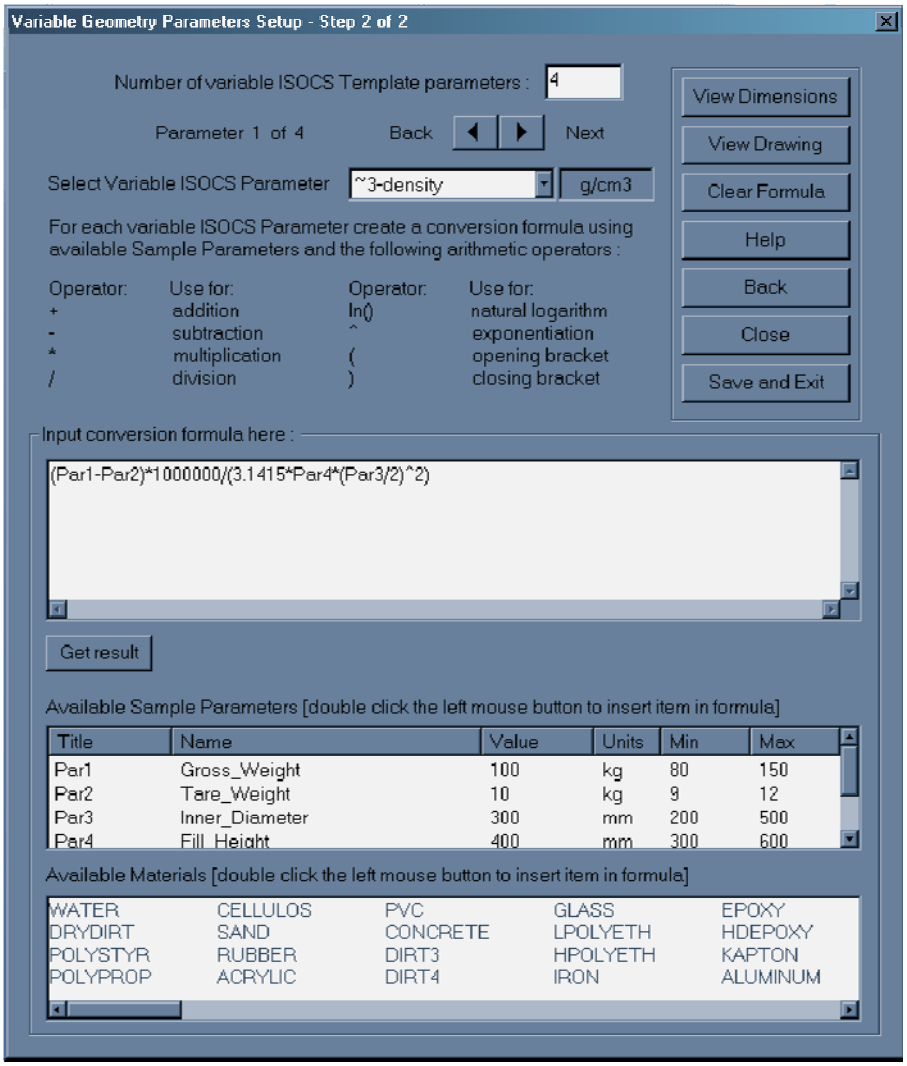


**Figure 2:** Variable geometry parameters setup

All information defined by the Expert using the Setup utility is stored in a special setup file, which can then be used by the Field ISOCS Analysis application to analyze the measured spectrum.

## 3. Field ISOCS Analysis application

The Field ISOCS Analysis application is used to perform the spectrum analysis based on the information, which is stored in the setup file created by the Field ISOCS Setup software. This application is to be used in the field by the Inspector and requires minimal user input.

The main screen of the Field ISOCS Analysis application is shown in Figure 3. Only a few simple steps are required to be completed by the Inspector prior to the spectrum analysis. The Inspector starts with selecting the setup file, which was prepared by the Expert. If the setup file contains information about the variable Sample parameters then a list of these parameters containing some default values will be available to the Inspector as shown in the screenshot below. The Inspector then

has the ability to modify any of these Sample parameters based on the information obtained in the field. After all Sample parameters were defined the Inspector selects a gamma spectrum, which needs to be analyzed, and then the spectrum analysis is performed automatically.

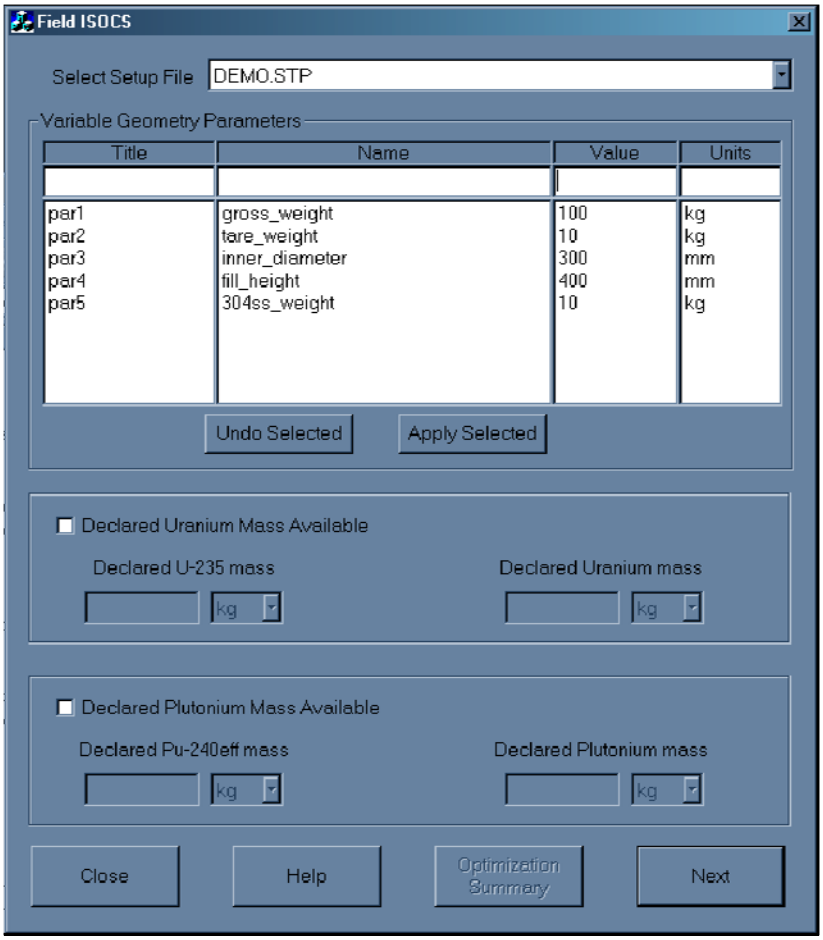


**Figure 3:** Field ISOCS Analysis application (main screen)

During the spectrum analysis the Field ISOCS Analysis software will use information from the selected setup file to automatically update parameters in the ISOCS geometry file based on the pre-defined formulae created by the Expert and new Sample parameters values entered by the Inspector. This eliminates the manual calculation and data transfer steps, which are utilized in the current measurement approach. There is no other input required from the Inspector, and the rest of the analysis process proceeds automatically. Following the geometry file update, the spectrum efficiency calibration is performed and the analysis results are generated based on the analysis steps, which were defined by the Expert in the ASF file. The analysis results are reported by using a customizable report template, which is also prepared by the Expert.

If the selected setup file contains additional information defined by the Expert, which requires the application to run geometry optimization prior to performing the efficiency calibration, then the optimization process will automatically proceed with no input required from the Inspector.

The overall Expert-Inspector spectrum analysis concept utilizing Field ISOCS is illustrated by the flowchart shown in Figure 4 below.

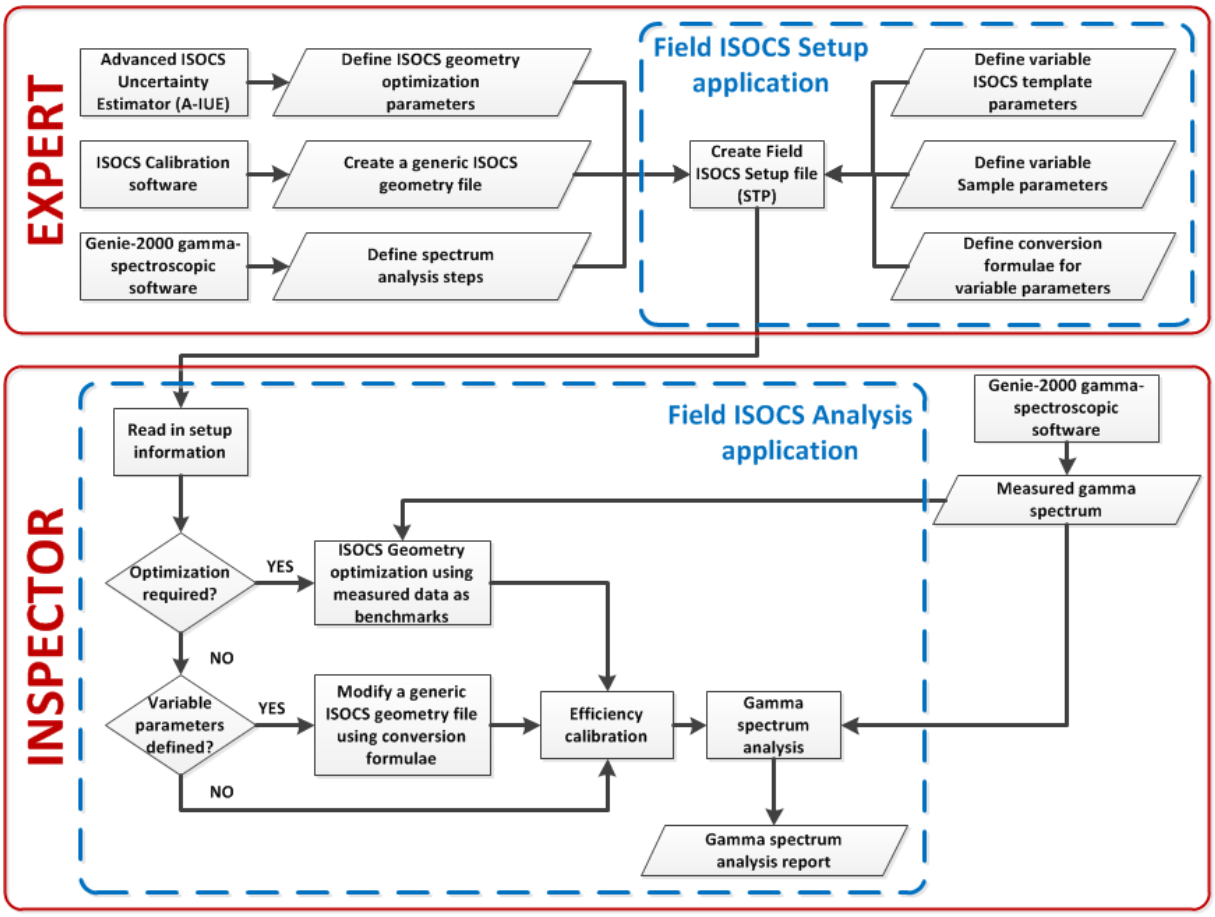


**Figure 4:** Expert-Operator concept for spectrum analysis

After the spectrum analysis is complete the report file will be automatically generated and presented to the Inspector.

## 4. Summary

The proposed Filed ISOCS application will consist of two separate utilities – Field ISOCS Setup and Field ISOCS Analysis applications.

The Field ISOCS Setup Application provides a tool for the Expert to setup the analysis procedure in advance for a given measurement scenario. All parameters of the analysis procedure are saved in a special setup file, which is then used by the Field ISOCS Analysis application. The Field ISOCS Analysis application is used by the Inspector in the field. It utilizes information contained in the setup file to perform the analysis of a measured gamma-spectrum.
Depending on the analysis setup the Field ISOCS application can be used to perform the following spectrum analysis routines:

- Running a simple ISOCS analysis by applying a pre-defined Genie-2000 Analysis Sequence File to the measured spectrum. This employs running any ASF defined by the Expert on any gamma-spectrum acquired by the Inspector. This analysis procedure is equivalent to simply running an analysis sequence from within Genie-2000 gamma-spectroscopic software.

- Running an ISOCS analysis for the known counting geometry having several variable Sample parameters. In this case the Inspector will be able to update the default values for each of the individual variable Sample parameters, which will then be used to update the generic ISOCS geometry information originally defined by the Expert for a given measurement scenario. The

software will then automatically perform the spectrum efficiency calibration and run a set of predefined analysis steps to generate a final result.

- If necessary, the analysis steps may also include geometry optimization using data retrieved directly from the measured spectrum. This will require creating a corresponding A-IUE project file by the Expert, which can be done in advance before the measurement campaign starts. The geometry optimization will be run automatically and will result in creation of an optimized efficiency calibration. The optimized efficiency data will then be loaded directly in the acquired gamma-spectrum and used to generate a final result.

The Field ISOCS application is designed to fully support the Expert-Inspector concept, which is often utilized by IAEA for measurements campaigns. It allows optimizing efforts for the Expert and Inspector, when an expert performs in-house equipment setup and calibration, while an inspector uses this equipment to conduct in-field measurement, performs spectrum analysis and generates results for NDA verification based on the input provided by the setup files. Some of the features introduced in the Field ISOCS software, such as an automatic spectrum analysis and ISOCS geometry optimization, as well as the ability to pre-define variable Sample parameters and their relationship with the geometry parameters in the ISOCS template, improve the accuracy of the results, make the user induced errors less likely to happen and also significantly reduce the overall time required to perform the spectrum analysis.

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# New and Emergent Information Technologies for Safeguards

Karl Horak, Sandia National Laboratories
Mailstop 1371, PO Box 5800
Albuquerque, NM 87185
E-mail: kehorak@sandia.gov

## Abstract:

Data regarding the acquisition of knowledge, equipment, and material for non-peaceful nuclear purposes can have many representations. Traditional relational databases are now being supplemented with newer graphical databases, many available as open-source software. In another domain, no computer can yet surpass the human brain for visual pattern detection. Outputting computational results in a comprehensible format can be critical to communicating findings. Open-source tools and software provide a number of applicable visualization methods. Finally, crowdsourcing is becoming an accepted mechanism for problem solving by enlisting the public at large. Recent exercises by the U.S. Department of State and Defense Advanced Research Projects Agency (DARPA) show promise in applying crowdsourcing to arms control and treaty compliance issues. In this paper the author will describe a number of open-source tools and methodologies for the collection and analysis of safeguards information. Focus will be on very recent trends and research.

**Keywords:** information technology, safeguards, database, visualization, crowdsourcing

## 1. Introduction

Information Science is a burgeoning field, no less in the safeguards domain than anywhere else. Keeping up with the pace of innovation seems a losing battle, but there are ways to lessen the cognitive load. In this paper the author would like lend a helping hand by exploring some of the new information technologies (IT) that are arriving on the scene daily. Three areas of interest have been selected out of the mountain of possibilities: graphical databases, visualization tools, and crowdsourcing. The result is an eclectic sampling of information technologies with an eye towards some that may have escaped the notice of the safeguards community.

As this article goes to press, it is already to some degree obsolete. IT projects that the author finished less than a year ago must still be updated frequently to keep pace with new versions of operating systems and browsers [1]. New data streams are constantly being discovered or old ones reworked with new tools to aid in the representation of information. Even though specific technologies mentioned here may become obsolete, it is hoped that this summary will encourage the reader to explore some of these tools and stay abreast of those areas of interest to themselves.

## 2. Methods

Before going into the details of selected new and emergent technologies, it would be useful to review the personal methods that the author uses to discover and track useful IT.

A first suggestion is to follow a carefully cultivated cadre of experts on Twitter [2]. In the sections that follow, the Twitter account names of relevant experts will be given parenthetically. The leading '@' sign indicates a twitter "handle" or account name. These tweets are public and can be read by anyone by using a URL in the form http://twitter.com/username, where username is the Twitter handle minus the leading '@' sign.

A recommended tool for managing Twitter followers is to use software such as TweetDeck [3] to manage them along with stored queries for topics of interest. A registered Twitter account is required, but in no way does that obligate one to actually publish any tweets.

A second strategy is to use social media like Twitter, Facebook and LinkedIn to ask specific questions and obtain guidance. The hashtag #lazyweb has become a keyword that indicates that one is broadcasting a question to the entire online community in the hopes of getting an answer without having to do a huge amount of research. Surprisingly, this works very well; experts are often willing to help out because it enhances their online reputation or provides other incentives. Some formal commercial systems such as Experts Exchange [4] promote this Q&A dialog by allowing answers to be rated and experts to gain "points" on a leaderboard.

It is completely acceptable to "unfollow" people or groups if their information is not relevant or useful. It can be beneficial to read first without contributing ("lurking") to learn logistics and basic etiquette of different social media platforms [5].

A third method is to regularly visit websites that aggregate current material on a topic of interest. RSS (Really Simple Syndication) or Atom feeds can be created to monitor this collection. In many cases a domain expert will use a service like paper.li [6] to automatically generate a weekly or daily online news summary. These can be gold mines of information about new services, software, and tools.

Finally, in all seriousness, keep your "digital natives" close at hand. These are generally considered someone born after the introduction of digital technology and through early interaction, have a greater understanding. One can learn a great deal from having a short visit with digital natives of any age because they make daily use of such a wide range of information tools and platforms.

## 3. Graphical Databases

SQL has long been the mainstay of database management. Limitations to the SQL and underlying relational database management systems (RDBMS) are poor support for sparse data and the necessity for complex schemas for complex relational models [7].

In recent years alternatives to RDBMS have begun appearing under the egis of NoSQL. Among these are object databases, RDF (resource description framework) databases, key-value stores, document stores, and graph databases. Most focus on scaling for large data sets. Graph databases focus on data structure.

A graph database differs from a RDBMS in that it provides index-free adjacency. Each item points directly to any adjacent elements and therefore can efficiently represent nodes, edges, and other properties of a network graph. It should be noted that graphs may be stored in any database management system, but efficiency and performance will be reduced when carrying out certain common operations.

Being able to quickly and efficiently manage network information is useful for safeguards applications because many problems are in fact network analyses. Acquisition path analysis, proliferation networks, social network applications, transportation routing networks, process flow models, many geospatial problems, and some types of semantic networks all can benefit from the use of a graph database architecture.

A recent ranking of graph database engines [8] places these five as the top contenders.

| |
|---|
| Neo4j [9] |
| OrientDB [10] |
| DEX [11] |
| InfoGrid [12] |
| InfiniteGraph [13] |

**Table 1. Graph Database Management Systems**

Neo4j is the significant leader in this field. The base installation uses the Cypher query language and a web-based console (Figure 1). Neo4j has drivers for REST, Java, Rails, Ruby, PHP, .NET, Python and others, allowing developers to work in whatever environment they are comfortable with.

```
Graph Setup:
create (orval{id:'1', name:'Orval', type:'BeerBrand'}),
(duvel{id:'2', name:'Duvel', type:'BeerBrand'}),
(heksemiebier{id:'3', name:'Heksemiebier', type:'BeerBrand'}),
(wilderengoud{id:'4', name:'Wilderen Goud', type:'BeerBrand'}),
(andorval{id:'10', name:'Abdij Notre-Dame d Orval', type:'Brewery'}),
(moortgat{id:'11', name:'Brouwerij Duvel Moortgat', type:'Brewery'}),
(wilderen{id:'12', name:'Brouwerij Wilderen', type:'Brewery'}),
(achilles{id:'13', name:'Microbrouwerij Achilles', type:'Brewery'}),
(hogegisting{id:'20', name:'Hoge gisting', type:'Beertype'}),
(trappist{id:'21', name:'Trappist', type:'Beertype'}),
(sixtwo{id:'30', name:'6.2', type:'AlcoholPercentage'}),
(eightfive{id:'31', name:'8.5', type:'AlcoholPercentage'}),
duvel-[:IS_A]->hogegisting,
duvel-[:ALCOHOL_PERCENTAGE]->eightfive,
duvel-[:BREWED_BY]->moortgat,
orval-[:IS_A]->trappist,
orval-[:ALCOHOL_PERCENTAGE]->sixtwo,
orval-[:BREWED_BY]->andorval,
heksemiebier-[:IS_A]->hogegisting,
heksemiebier-[:ALCOHOL_PERCENTAGE]->sixtwo,
heksemiebier-[:BREWED_BY]->achilles,
wilderengoud-[:IS_A]->hogegisting,
wilderengoud-[:ALCOHOL_PERCENTAGE]->sixtwo,
wilderengoud-[:BREWED_BY]->wilderen


Query:
START duvel=node:node_auto_index(name='Duvel'), orval=node:node_auto_index(name='Orval')
MATCH p = AllshortestPaths( duvel-[*]-orval )
RETURN EXTRACT(x in p : COALESCE(x.name?,TYPE(x))) as path, p;
```

**Figure 1. The Neo4j console.**

A simple output graph from the system is shown in Figure 2, but Neo4j is known for its speed in processing large datasets (traverses 1,000,000+ relationships per second on commodity hardware [7]).

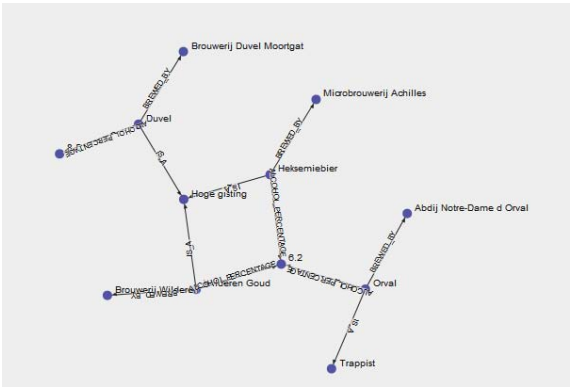


**Figure 2. A simple Neo4j output graph**

One interesting system not in Table 1 is FlockDB, the graph database engine that is used by Twitter. While FlockDB is optimized for very large adjacency lists, it is not particularly well suited to graph traversal operations [14]. This would make it an excellent choice for social media applications but a poor choice for routing (shortest path) calculations and semantic network problems.

(@ChrisDiehl, @Aurelius, @neo4j, @dgleich, @arnicas, @jeffery_heer, @gutelius)

## 4. Visualization Tools

Visualizations are becoming increasingly important for a variety of reasons:

- Humans have evolved to be visually perceptive – this is often the best route to comprehension.
- The complexity of many data sets is so high and the volume of information is so vast that skillful visualizations are necessary for understanding.
- Gaps in data can more easily be spotted in a coherent visualization rather than tabular or textual formats.
- Decision makers must often render a time-critical judgment without digesting the full and complete documentation.
- Visualizations help one actually see the statistics involved in a metric.

Of the many visualization tools available, in this paper we will look at Gephi for graph visualization, GeoSafeguards for text-to-map conversion, some general text-based tools, and a technology stack for augmented reality.

Gephi [15] is an open-source graphing system capable of handling very large networks and providing a large array of powerful features. Gephi permits filtering, viewing ego-centric networks, calculating various metrics, and using colors, sizes, and widths to display different quantities.

Figure 3 illustrates a Gephi graph of Sandia co-authorship of 271 safeguards papers presented at INMM since 1974. Nodes are placed by means of a force-based algorithm to spread them apart. Colors represent clusters determined by attribute commonality, in this case, author's organization. Sandia National Laboratories is the prominent central node due to the selection criteria used in creating the data set. Line width indicates the strength of connection (in terms of number of papers co-authored) between nodes. Although details are not visible, the general form of the graph illustrates the breadth and interconnectedness of the safeguards community.

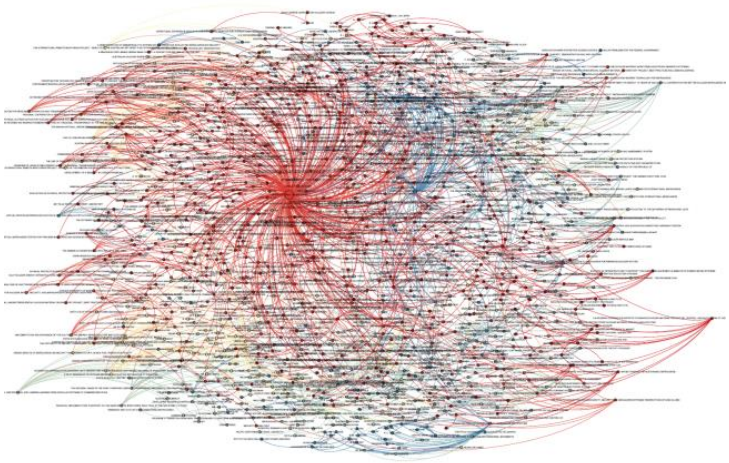

**Figure 3. Gephi co-authorship graph**

Geospatial relationships are best understood when viewed as maps. GeoSafeguards [16] is a browser-based tool for generating digital maps based on placenames found in collections of online documents. GeoSafeguards was developed by Sandia National Laboratories in 2012. The ability to rapidly create maps from text documents means non-GIS analysts can now easily visualize the geographical nature of their research. Figure 4 shows a map generated from a collection of online news articles about the 2010 toxic flood in Hungary. Not only are placenames automatically extracted from the documents, they are geolocated, and the context within which the placename is mentioned can be viewed by clicking on the map tack. Different map layers can be used as the base map and a set of useful measurement tools is provided.

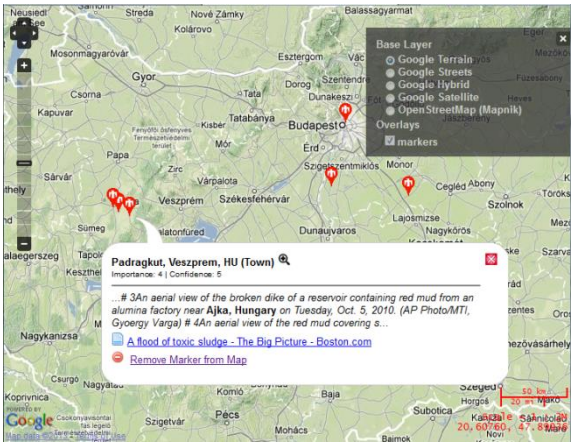


**Figure 4. GeoSafeguards map**

A number of useful tools are available to analyse text. Wordle [17] is an online word cloud generator. It rapidly creates word clouds in a variety of styles from copied text or a specified URL. Figure 5 is an example based on the text of this paper. Tagxedo [18] and TagCrowd [19] have similar capabilities but with differing format styles and word filtering. Word clouds use font size and color to visualize word frequency within a document, but the spatial relationships are random. This has the effect of generating new associations in the mind of the viewer as they examine novel patterns of words that are emphasized based on frequency of use, not semantic relationships.


**Figure 5. Wordle word cloud of this article**

IBM's Many Eyes online tool [20] provides a large number of visualizations for text along with other capabilities for examining numeric data. Figure 6 shows a sample word tree based on this article. A word tree is a visualization of a concordance or, technically speaking, a suffix tree. Word trees enable one to rapidly find all the instances of a term or phrase in a document and view its context.

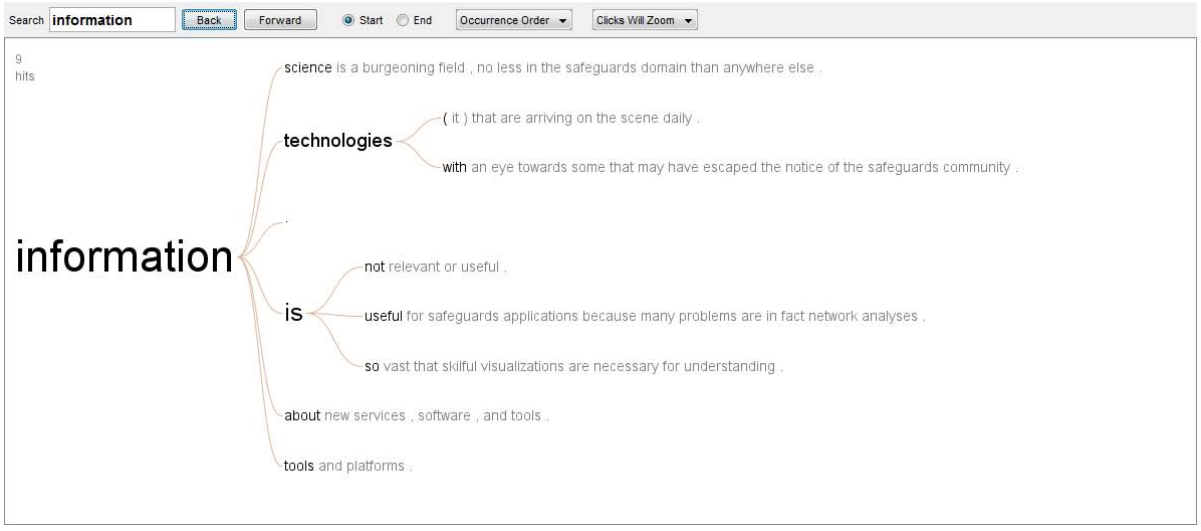


**Figure 6. Many Eyes word tree**

(@themisoproject, @viegasf, @wattenberg, @ushahidi, @parry_joe, @geosafeguards)

Augmented reality (AR) is a technology that overlays a live view of a real-world environment with additional digital content. This differs from virtual reality (VR) where scenes are entirely artificial. Simple systems that link text data to smartphone camera location and direction of view (Figure 7) have been available for several years [21].

**Figure 7. A simple AR example**

More complex interactive AR applications are now possible. Wearable systems like Google Glass are approaching commercial release [22]. Sandia has made use of the following technology stack for building AR applications on Android mobile devices (Table 2).

| | |
|---|---|
| Android SDK, ADT and NDK [23] | Software Development Kit, Android Development Tools, Native Development Kit |
| Blender [24] | 3D graphics package |
| Unity [25] | 3D gaming engine |
| Vuforia [26] | AR engine |

**Table 2. AR technology stack**

Blender is a powerful 3D graphics package, which can import models from a wide variety of sources—Colada, Google Sketchup, and others. In our experience, being able to make use of legacy 3D models and in-house graphics talent is important to making effective use of existing staff and flattening the learning curve.

Unity is a commercial 3D gaming engine capable of rapidly rendering complex models even on early generation smart phones. Unity imports Blender models seamlessly. Vuforia is a Qualcomm product that links Unity models with reference frameworks in the mobile camera's field of view. AR requires a frame of reference for proper display of an associated 3D model. It is necessary to have staff capable of working with all aspects of the technologies involved.

AR applications may be installed on mobile devices and used for training, for demonstrating mechanical procedures, for visualizing complex data sets, and for adding additional information on top of existing documents and illustrations (Figure 8).


**Figure 8. AR view of the digital 3D model atop a reference image**

AR models can be created that respond to touchscreen operations as well as events within the camera field of view. This provides a rich environment for interaction with combinations of real-world and virtual content.

(@vuforia, @Unity3D, @Blender3D, @projectglass)

## 5. Crowdsourcing

Van Ess [27] defines crowdsourcing as "channelling the experts' desire to solve a problem and then … sharing the answer." Typical areas where crowdsourcing have found application are "... problems that bug people, things that make people feel good about themselves, projects that tap into niche knowledge of proud experts, subjects that people find sympathetic or any form of injustice." Crowdsourcing can take many forms (Table 3).

| | |
|---|---|
| Crowdvoting | Tallying online votes, "likes," or recommendations |
| Crowdfunding | Requesting donations to generate startup funds, for disaster relief, etc. |
| Wisdom of the crowd | Using data from a very large set of online contributors to estimate an answer. |
| Challenges (crowd prize contests) | Public challenges where a prize is offered to the first or best solution to a stated problem. |
| Implicit crowdsourcing | Crowdsourcing by means of some other task, for example, game playing, ReCaptcha, Google searches, etc. |

**Table 3. Crowdsourcing techniques**

Crowdvoting and crowdfunding, for example, KickStarter [28], appear to have limited application to safeguards. Other types of crowdsourcing activities, though, are garnering much more attention. Rose Gottemoeller, Acting Under Secretary for Arms Control and International Security has been spearheading the U.S. Department of State's efforts [29]. Their 2012 TAG Challenge [30] resulted in an MIT-affiliated team locating three of the five target individuals who were spread out among New York, Washington, Stockholm, London, and Bratislava [31]. Within the U.S. government, NASA and DARPA are also heavily involved in crowdsourcing projects [32].

Crowdsourcing challenges have met with such success that it is spawning its own industry with companies like Innocentive hosting dozens of challenges at any one time [32]. Amazon's Mechanical Turk program [33] lets experts pose tasks that require human intelligence and workers anywhere in the world may perform them. Difficult pattern recognition tasks are particularly well suited to this solution architecture.

(@DataKind, @amazonmturk, @InnoCentive, @fei_innovation, @GBFAI, @jenngustetic, @TheIdeasEconomy, @simon_crowd, @karinahomme, @agjs, @womenovation, @_innovation, @IM_Innovation, @CrowdsourcingKC, @Crowdsourcerer, @the_idea_agency, @Gottemoeller)

## 6. Conclusions

Safeguards-relevant data is growing in absolute volume, in number of potential formats, and in complexity. Staying abreast of information technologies that can assist in managing this growth is a daunting task.

Standard computer tools like operating systems, word processors and spreadsheets are changing, evolving, and growing more powerful even while the safeguards information toolkit does as well. To add another layer of information tools from the Web at large risks placing safeguards professionals under too large a cognitive burden. This risk must be balanced against the value these new technologies can bring to the domain.

Fortunately, some of the new and emergent technologies can help reduce the impact of information overload by connecting us with experts, allowing us to use their filters to preselect tools that have a higher likelihood of paying dividends if adopted.

(@mashable, @timoreilly, @4ManagingChange, @Changely, @Manage2change)

## 7. Acknowledgements

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SAND 2013-3839C.

## 8. Legal matters

### 8.1. Privacy regulations and protection of personal data

I agree that ESARDA may print my name/contact data/photograph/article in the ESARDA Bulletin/Symposium proceedings or any other ESARDA publications and when necessary for any other purposes connected with ESARDA activities.

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The author agrees that submission of an article automatically authorises ESARDA to publish the work/article in whole or in part in all ESARDA publications – the bulletin, meeting proceedings, and on the website.

The author declares that their work/article is original and not a violation or infringement of any existing copyright.

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# What Future Role for ESARDA as a Reference European Technical Think Tank?

**Michel Richard,**

Independent expert[1],
Email : micheljrichard@orange.fr

***Abstract:***

*The objectives of ESARDA aim « to advance and harmonize research and development in the area of nuclear safeguards », within international safeguards and to ensure mutual exchange of information and technical assistance to promote the advancement of safeguards, enhancing the efficiency of systems and measures, to investigate on how new techniques can be developed and implemented, to have an educational role. The RG2010 has recognized that these objectives are still valid. ESARDA is a unique European forum for the exchange of information and ideas which bring together all those involved in safeguards: nuclear facility operators, safeguards authorities and persons engaged in research and development. ESARDA has 43 years of existence, the Association. It has grown from two parties to 27 parties from 14 EU countries, plus 3 associated members and 9 individual members and 1 observer. During this time, the European and international context in the field of nuclear safeguards, non-proliferation, arms control and security has dramatically changed. The Association has adapted permanently to the evolution. With the implementation of the recommendations put in place after the 2000 and 2010 reflection groups which had given ESARDA the tools to deal with the current and future security challenges. ESARDA has all the necessary competences to investigate the technical aspects of issues on global security.*

*The author examine and propose ways on how ESARDA, as a reference European technical think tank, building on its expertise base, its organization and methods of work could improve its role of advisor of the EU member States, the EU authorities and the IAEA on difficult question and improve its links with other sister organizations as inter alia, INMM, VERTIC, SIPRI,…*

**Keywords:** ESARDA, future, think-tank, non-proliferation, Arms control, safeguards

## 1. Introduction[2].

As pointed up the 2010 Reflection Group, ESARDA's main objective as stated in the1969 agreement:

*"**To advance and harmonise research and development in the area of nuclear safeguards**[3] », within international safeguards and to ensure mutual exchange of information and technical assistance to promote the advancement of safeguards, enhancing the efficiency of systems and measures, to investigate on how new techniques can be developed and implemented, to have an educational role and to reach the general public, in particular, those interested in safeguards".*

Has been reach and is still valid. The core business of ESARDA is R&D activities and the implementation of the results by plant operators and safeguards authorities either national, European or the IAEA. ESARDA also endeavor to fulfill an information and educational role aiming to reach all

[1] ARMIR Global Security Network/Former advisor for international issues of the Director of Security and Non Proliferation CEA

[2] The ideas set for in this paper draw from the ESARDA reflection group reports: 1993, 2000, 2010, from the report of the ESARDA WGs to the board and from a 2010 non-published paper from Bruno Pellaud to the RG 2010.

those interested in safeguards and more generally in nuclear nonproliferation and security through the bulletin, the symposia and the non-proliferation course.

ESARDA and its working groups enjoy a very good reputation among scientists, engineers, authorities and operators of the EU and beyond. Tights relations have been established with the IAEA Safeguards Department, Safety and Security Department and Think-tanks, institutes and Academia: INMM, SIPRI, VERTIC, Carnegie, King College, Liege, Hamburg, Non-European organizations (US, Japan, Korea,..), based on the high quality of the work in the working groups.

In the future, after taking stock of the past decades, may be 2013 will be considered as a milestone in the life of ESARDA. The recommendations put forward by the reflection groups 2000 and 2010 have been almost implemented and ESARDA is drawing benefit of the managerial changes and the extension of the scope of its activities. Beyond the IAEA and EURATOM safeguards ESARDA thanks to the expertise of its members, is able to deal with successfully, "politically and technically sensitive issues" complex and difficult to handle about which the views of ESARDA members could somehow differ as nuclear security, proliferation resistance, illicit trafficking, nuclear terrorism prevention, disarmament verification and so on. ESARDA working groups draw from these issues useful material to be used by members and national or international bodies

The objective of this paper is to give food for thought to a next reflection group to initiate thinking of ESARDA parties, of the Steering and the Board on the future "strategic" stand of ESARDA on all these issues and on its relationships with regards to members, national and European authorities, international organisations, on top of them the IAEA and think-tanks.

Some ideas could seem iconoclastic but looking back, it was also the case of several proposals dealt with in the 2000 reflection group report and in the symposia which took place at that time. Some of these proposals have been put forward by the 2010 reflection group and are now applied without any problem. Relying on my personal experience of ESARDA since 1998 I think time is ripe to initiate thinking on ways and means to give ESARDA more visibility and more weight as a as a reference European technical Think-Tank, based on the quality and diversity of its members.

## 2. ESARDA: an on-going adaptation to the world.

Since its creation in 1969 and during his 44 years of life, the European Safeguards Research and Development Association, ESARDA has permanently adapted its management, objectives and activities to closely fit to the evolution of IAEA, EURATOM and the international context. The two main milestones were:

- First, the implementation of comprehensive safeguards consecutive of the entry into force of the NPT for EURATOM members (70's) which led to the birth of ESARDA to control the process;

- The implementation of strengthened comprehensive safeguards (90's) and the entry into force of the Additional Protocol (2004) for the members of the Union;

At the same time and in the same movement ESARDA gradually expanded the scope of its activities, taking into account and integrating into the working groups activities, the evolution of the international and European context regarding nuclear security issues in the broad sense of the term, issues stemming from crises that occurred since the early 90s inter alia the following ones:

- Collapse the Soviet Union, the proliferation crises in North Korea, Iran, Libya, Syria, Pakistan,

- Unveiling of undercover networks of acquisition of sensitive technology such as the AQ Khan's and after September 11 2001,

- Strengthening of nuclear security in response to threats from terrorist groups,

- Verification of nuclear disarmament: elimination of nuclear weapons, elimination of excess fissile materials no longer needed for defines purpose, verification of a treaty banning the production of fissile material for nuclear weapons (Cut-off treaty) or preparation of On-Site Inspections (OSI) of the Comprehensive Test Ban Treaty (CTBT) when it will enter into force.

This permanent adaptation has been achieved through the of the members, the exchange during the annual symposia but especially through the reports of the working groups to the Board and the recommendation of the reflection groups which have been set up in the past

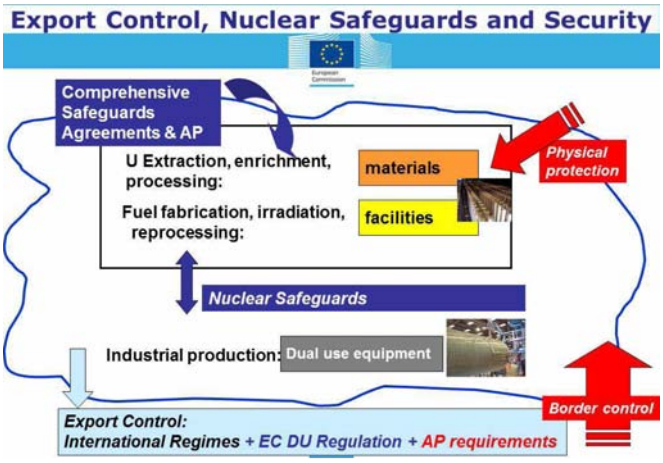
The critical technical and managerial evolution of ESARDA initiated by the 1993 reflection group recommendation started at the bascule of the centuries, in fact at the time of the Helsinki conference in 1998. It is now achieved and its outcomes could be appreciated.

Working Groups are now optimized and rationalized. Some have disappeared groups as the topic they dealt with were important at a time but are no longer relevant or have been merged with other working group's one:

- Integrated Safeguards working group;
- Nuclear Materials Accounting and Control working group;
- LEU Conversion and Fuel Fabrication Plants working group;
- MOX Fuel Fabrication Plants working group;
- Back-end of Fuel Cycle working group;

All their topics they dealt with are managed now by the Implementation of Safeguards working group. Some other working groups have been created from the Verification Technologies and Methodologies WG (VTM WG) as:

- Novel Approaches/Novel Technologies WG and,
- Export Control WG
  (credit Filippo Sevini/JRC)




The Training and Knowledge Management working group is rather new and aims to fulfil the information and educational role of ESARDA jointly with the Editorial Committee (Edom) which manages the annual course, the bulletin and cooperation with academia at a time when training and education in safeguards, non-proliferation and security becomes more and more important.

The ESARDA course on Nuclear Safeguards and Non Proliferation managed by the JRC and the Editorial Committee and the TKM WG is a very successful and visible activity which contributes significantly to fulfil the objective of the Association and other sessions have been and will be set up in other EU members (Sweden) and abroad (Malaysia).

The three "historic" disciplines oriented working groups:

- Destructive Assays
- Non Destructive Assays
- Containment & Surveillance

Have kept their structure but have dramatically increased their field; For example, inter alia, DA WG which address new challenges in measurement problems arising from safeguards and in related areas and new measurement techniques, emphasize convergence of nuclear safeguards, nuclear forensics and nuclear security and Support activities for the development and improvement of methods for determination of nuclear signatures in environmental and special samples.

Moreover as it could pointed out, browsing the agenda of ESARDA meetings that working groups are working closely together, bringing their competences to tackle complex issues as NA/NT works with VTM, DA and NDA on Forensics, nuclear disarmament verification, arms control, stand-off detection technology.

This trend of the working group's cooperation allows defining a limited number of horizontal themes which would constitute the objectives of ESARDA (as it is already the case for safeguards).

## 3. ESARDA to day

### Membership:

Now ESARDA has 27 parties representing 14 EU member states + the Join Research Centre, 3 associated members, 9 individual members and 1 Observer (DG ENER).

The issue of the membership has not been really taken up by the 2010 Reflection Group which make only a few remarks on the extension of the membership and the individual members (cf. infra) . This question has been also discussed in the paper from Bruno Pellaud (ref. 2).

What is noteworthy is the composition of the membership of ESARDA though somewhat heterogeneous which includes commercial operators covering the whole nuclear fuel cycle, governmental offices, inspectorate authorities, R&D organizations including the DOE laboratories of Oak Ridge as associated member, and since a couple of years academic institutions and individual members

The broadening of the membership to academic institutions and individual members are an important fact to take into account for the future stand of ESARDA. Membership of academic institution convey the objective of ESARDA to have an educational role aiming to promote nuclear safeguards, non-proliferation and security and to benefit of academic R & D in the working groups..

### Extension of membership?

In the EU and associated countries, the horizontal extension of the membership is almost completed. ESARDA should look to get new members vertically in the countries already represented in the Association. Other national authorities in the new EU member states should be approached. Should ESARDA look for associated member abroad? ESARDA has already tight links with INMM and through INMM with the DOE laboratories. May be BACC? Or should ESARDA propose an associated membership to Japanese and Korean organizations. This worth to be discuss in a future reflection group.

### Individual members?

The admission of individual members has been proposed by the Reflection Group 2000. In 2013, there are only 7 individual members. But this number will probably increase significantly in the coming years as old members will retire and wish to keep an active role in the association. For ESARDA this could be both a chance and an administrative burden. An administrative burden as ESARDA secretariat will not be able to manage numerous individual members + the current institutional members. But it will be also a chance because more individual members would increase the leverage of ESARDA in its promotion and communication activities. They will also certainly play a constructive role in the life of the working groups. The target should not be a crowd, but altogether about a few dozen of qualified and motivated persons, even from within ESARDA member-organizations. The

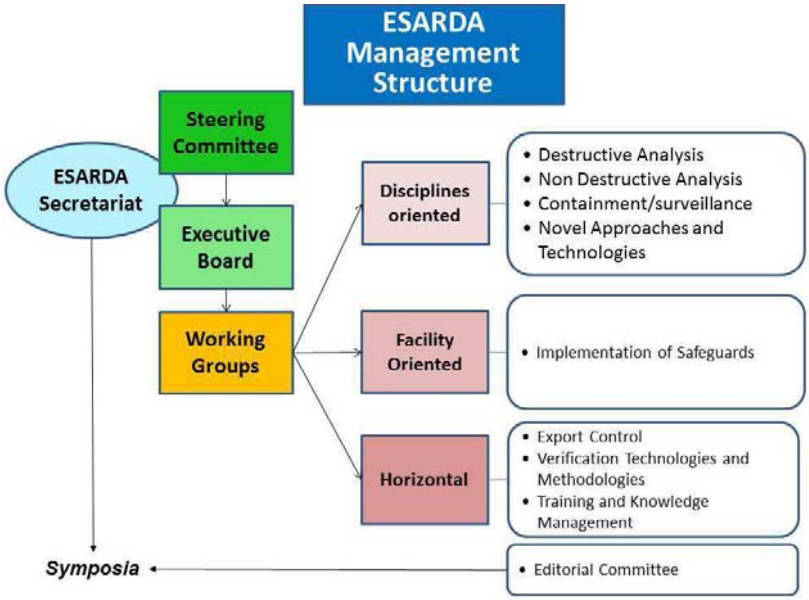
status of such individual members would be comparable to that of Honorary Members (or to the granting of Merit Awards) for persons having contributed to the objectives of ESARDA. Whatever, a future reflection group would have to dedicate some attention to these 2 questions as they are important for a future stand of ESARDA.

One point which worth drawing the attention of the Board and the Steering prior to any action on the extension of the membership and may be to submit it to a future reflection group, is to get a better knowledge of the actual participation to ESARDA life symposia, working group meetings bulletin articles. Who and how many participants are actively involved in the life of ESARDA? 250, 300? The reflection group 2010 tried to move forward on this issue but did not succeed to have a clear view of the ESARDA life's contributors.

### Working group:

**ESARDA has 9 working groups:**

| Discipline oriented | Horizontal |
|---|---|
| Destructive Assays | Implementation of Safeguards (Facility oriented) |
| Non Destructive Assays | Verification Technologies & Methodologies |
| Containment & Surveillance | Training & Knowledge Management |
| Novel Approaches/Novel Technologies | EdComm |
| | Export control WG |




## 4. ESARDA and other Think tanks

Wikipedia gives the definition of think tank as the following: policy institute often termed "think tank" is an organization that performs research and advocacy concerning topics such as social policy, political strategy, economics, military, technology, and culture. Most policy institutes are non-profit organizations, which some countries such as the United States and Canada provide with tax exempt status. Other think tanks are funded by governments, advocacy groups, or businesses, or derive revenue from consulting or research work related to their projects.

The question is: should ESARDA become a think-tank and join the European Network and what would be the benefit for the Association.

The answer is: ***ESARDA is already a think tank***.

In the world and in the EU there are many Institutes, Foundations, Associations, Think-tanks which deal more on less on the same topics than ESARDA, but among them ESARDA enjoy a special position:

ESARDA is the European voice on nuclear verification now taken in a broad sense. Due to the wide composition of its membership: governmental offices, R&D organizations and commercial operators, inspectorate, universities; in and outside the EU and the skill of participants: engineers, layers, scientific researchers, IAEA EURATOM or national inspectors, university professors and so on, ESARDA held a unique position among the European think-tanks but ESARDA possibility of action is limited due to its status.

To give more visibility and dedicate more resources to topics which became important ones with regards to the IAEA and EURATOM safeguards and the global security context (i.e. disarmament verification issues, nuclear security, terrorism acts prevention, etc.., worth to have ESARDA voicing on, should its status and its management be modified?

Should ESARDA enhance its role as part of EU policies?

Let examine the status of several think-tanks the objectives and activities of which are close to ESARDA:

**INMM:**

INMM is the US sister of ESARDA and the two organisations have developed along the years tight links through which joint projects inter alia the periodic joint meeting INMM ESARDA dedicated to top issues or the Nuclear Safeguards and Non Proliferation Education and Training, initiatives by ESARDA, INMM and JRC. This cooperation has been materialized in a Letter of Intent (LOI, 10/2011) by which they recognize that the missions of INMM and ESARDA are similar and respond to the increased importance to international security of nuclear material management and safeguards due to recent events, take note of the on-going collaboration between INMM and ESARDA and plan several actions to strengthen relationships and improve collaboration.

Nevertheless INMM organisation and operation is rather different than ESARDA ones. INMM has a large membership of individual members. Besides its domestic chapters, INMM has set up a number of foreign chapters to attract foreign members, in the UK, Japan, Korea, Russia, Ukraine and in Vienna dedicated to IAEA staff members. INMM operations are managed by a permanent secretariat and has its own budget founded by the fees of the members, the contribution of organization and the receipt from events like the annual conference.

To what extent INMM constitutes a “model” for ESARDA is a point worthwhile to be discuss in a future Reflection Group.

In the EU some organisations cover more or less the same areas than ESARDA among them VERTIC (UK), SIPRI (SW), with the difference that they have also a standing organisation but does not benefit of the pool of expertise of ESARDA. Several of them participate actively to the activities of the working groups.

At this stage its worth saying a few words on the ***EU non-proliferation Consortium*** (http://www.nonproliferation.eu/ ). In July 2010 the Council of the European Union decided to create a network bringing together foreign policy institutions and research centres from across the EU "*to encourage political and security-related dialogue and the long-term discussion of measures to combat the proliferation of weapons of mass destruction (WMD) and their delivery systems*". The EU Non-Proliferation Consortium is managed jointly by four institutes in close cooperation with the representative of the High Representative of the Union for Foreign Affairs and Security Policy. The four institutes are

- Fondation pour la Recherche Stratégique in Paris,
- Peace Research Institute in Frankfurt (HSFK/ PRIF),
- International Institute for Strategic Studies (IISS) in London
- Stockholm International Peace Research Institute (SIPRI). T

The Consortium forms the core of a wider network of European non-proliferation "think tanks" and research centres closely associated with the activities of the Consortium.

*The participation of ESARDA to this network is a question which worth being discuss in the board and the steering*

## 5. Possible next step for ESARDA

The following ideas (which just commit the authors) should be considered as "food for thought for the Next Reflection Group

The Reflection Groups 2000 and 2010 have provided recommendations on the scope and operation of ESARDA, most of them have been implemented and have also provided recommendations on the organisation and management in the framework of the actual status of the association (except the strategic plans)[4]. The implementation of the RG 2000 recommendations have permitted a better operation of the association but find now some limitations with the broadening of the scope of WGs activities and the opening of new possibilities of cooperation with EU and non EU partners.

ESARDA's core business is on research and development in the nuclear security areas, including the coordination, cooperation and dissemination of these R&D activities. Coordinated research and dissemination actions cannot often be executed based solely on the internal funding of ESARDA partners, however engaged they may be to support the Association.

ESARDA governing bodies and a future reflection group should examine how to sensitise agencies on ESARDA R&D funding needs and how to be pro-active in coordinating efforts towards those needs: i.e., creating a task force to draft needs for coordinated research as well as lobbying near funding agencies[5]

**How to fund research projects?**

It should be discussed and arranged with the EC and members that ESARDA gains a special status for creating research networks and proposing research projects. For this purpose a special board or group should be created which reviews and approves projects to guarantee the needed quality of the research projects and proposals. The participation in these projects should not be limited to European members but it should also be allowed to have other partners e.g. from US (through INMM?) and East Asia (through TKM network?).

This raise the question of ESARDA funding: , like other Institutes or Association (INMM, VERTIC, SIPRI,…) having dedicated funding resources and a dedicated staff to manage projects?.

- Should ESARDA explore the possibility to have dedicated funding resources and a dedicated staff to manage projects like other Institutes or Association (INMM, VERTIC, SIPRI,…)?

- Or should ESARDA strive to develop inside the actual status?

There is yet no answer to this question. It’s up to a dedicated group to explore these tracks. It’s a long way. It has taken 15 years for ESARDA to go beyond the safeguards: Helsinki (1998) ⇨Bruges (2013)

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4 See the presentation of A. Rezniczek and G. Stein to the RG 2010: "ESARDA Reflection Group 2000", Objectives- Achievements Recommendations.

5 Recommendation from RG 2010.

# The detection of reactor antineutrinos for reactor core monitoring: an overview.

**Muriel Fallot**

SUBATECH, CNRS/IN2P3, Université de Nantes,
Ecole des Mines de Nantes, F-44307 Nantes, France
E-mail: fallot@subatech.in2p3.fr

***Abstract:***

*The field of applied neutrino physics has shown new developments in the last decade. The International Atomic Energy Agency (IAEA) has expressed its interest in the potentialities of antineutrino detection as a new tool for reactor monitoring and has created a dedicated ad-hoc Working Group in late 2010 to follow the associated Research and Development. Several research projects are on-going over the world either to build antineutrino detectors dedicated to reactor monitoring, either to search for and develop innovative detection techniques, or to simulate and study the characteristics of the antineutrino emission of actual and innovative nuclear reactor designs.*
*The European Safeguards Research and Development Association, ESARDA, has created in late 2010 a group devoted to Novel Approaches and Novel Technologies (NA/NT) allowing to create contacts between the research community and agencies. The ESARDA NA/NT working group has decided one year ago to create a sub-WG dedicated to the detection of antineutrinos. At this 35th ESARDA meeting, we propose to give an overview of the most recent progresses made in the field of antineutrino detection for reactor monitoring, including the actual possibilities and limitations of their detection and the status of various developments towards compact antineutrino detectors for reactor monitoring considered in perspective of the antineutrino emission from various reactor designs. We will then present the objectives of the ESARDA sub-WG devoted to the antineutrino probe.*

**Keywords:** safeguards, non-proliferation, antineutrino detection, nuclear reactors

## 1. Introduction

The research and Development (R&D) associated to reactor antineutrino detection is very lively. This 35$^{th}$ ESARDA annual meeting is an opportunity for the antineutrino community to meet in the frame of the NA/NT Working Group (WG) [1]. At this meeting, the proceedings of the last antineutrino detection ad-hoc WG of IAEA (Oct. 2011) will be presented to the attendees, giving the directions that were foreseen by then. After this first part, overview talks will focus on several topics of importance in the field: a review about each main actual detection techniques, a review about reactor simulations, and reviews making the links with other fields strongly connected to the topic: reactor antineutrino detection for fundamental neutrino physics, nuclear physics experiments for reactor antineutrino energy spectra and neutron detection techniques.
In these proceedings we will present the motivations for such a structure of the sub-WG meeting, explaining the context and presenting briefly the different topics of the talks.

Let us first recall briefly the principle of reactor monitoring with antineutrino detection.
Large quantities of antineutrinos are produced in the reactor due to beta decays of the fission products and about $10^{21}$ antineutrinos/s are emitted by a 1 GWe reactor core. The distribution of fission fragments depends on the fissile isotopes ($^{235}$U, $^{238}$U, $^{239}$Pu and $^{241}$Pu) and on the energy of the neutrons in the core. The released energy per fission, the average number of emitted antineutrinos and their average energy depend also directly on the fissile isotope that undergoes fission (see Table 1). Consequently, an antineutrino spectrum measured at a reactor will reflect the thermal power emitted by the core and its composition. Adding to these features the intrinsic properties of antineutrinos which are weakly interacting particles, impossible to shield, the antineutrino detection may then become an interesting tool for reactor monitoring.

| | $^{235}$U | $^{238}$U | $^{239}$Pu | $^{241}$Pu |
|---|---|---|---|---|
| Released energy per fission (MeV) | 201.7 | 205.0 | 210.0 | 212.4 |
| Mean energy of antineutrinos (MeV) | 1.46 | 1.56 | 1.32 | 1.44 |
| Number of antineutrinos per fission (E>1.8 MeV) | 5.58 (1.92) | 6.69 (2.38) | 5.09 (1.45) | 5.89 (1.83) |

**Table 1:** Differences in the $^{235}$U, $^{238}$U, $^{239}$Pu and $^{241}$Pu fission properties given in [2] and a calculation of P. Huber and Th. Schwetz [3].

Considering these properties, the IAEA asked to its member states to perform a sensitivity study. The agency organized several meetings between experts and inspectorates since 2003, and created in 2011 an Ad-Hoc Working Group devoted to the antineutrino detection. This WG meeting was associated to an Antineutrino Applied Physics Workshop [4] and a proceedings was written and circulated [5]. The content of the proceedings will be presented by T. Shea and A. Bernstein for the first time to the R&D community and to ESARDA and IAEA members at our next ESARDA NA/NT WG meeting on May 27. The proceedings give indications of directions for the research to come on the path to use antineutrino detection for reactor monitoring and potentially for safeguards.
An antineutrino detector should be as compact as possible, cheap, safe, unattendedly and remotely operated and the data analysis should be performed in a nearly automated way in order to be usable by inspectorates. The optimum situation of an antineutrino detector would be above ground jus outside a reactor building.
These requirements remain a challenge for physicists. A cubic meter size footprint for an antineutrino detector placed at about 25m from a reactor core is feasible, but one has up to now to add shieldings around the detector in order to get rid of cosmic ray or reactor induced background, and the shielding enlarges a lot the footprint. At the moment eliminating the shielding is not compatible with a detector located above ground. But as you will see in the following, the on-going developments show promising improvements in background rejection efficiency by an antineutrino target.

## 2. Main antineutrino detection techniques

The most common interaction used to detect reactor antineutrinos is the inverse beta decay (IBD) process on proton: antineutrino + p => $e^+$ + n (threshold: 1.8 MeV ). Up to now liquid scintillator targets were used as the standard detection technique, usually doped with Gd in order to capture the emitted neutron. The positron energy is directly related to the antineutrino energy. After a neutron capture, the excited Gd isotope emits a gamma cascade of total energy 8 MeV detected in a delayed coincidence (delayed signal) (in average 30 µs) with the positron signal (prompt signal). This technique was used extensively by numerous fundamental neutrino physics experiments. The quality of the liquid scintillator doped with Gd has been the object of an important R&D and the new generation reactor experiments Double Chooz, Daya Bay and Reno use this method [6-8]. But one could replace Gd with other nuclei exhibiting large neutron capture cross-sections such as $^6$Li and $^{10}$B. The advantages of these nuclei is that neutron capture gives birth to heavy nuclei which can be distinguished from gamma rays through Pulse Shape Discrimination techniques (PSD): n + $^6$Li => $\alpha$ + $^3$H and n + $^{10}$B => $^7$Li + $\alpha$ (6%) and $^7$Li + $\alpha$ + 0.48 MeV (94 %). These neutron captures occur for neutrons less thermalized than in the case of neutron capture on Gd isotopes leading to a shorter path of the neutron in the detector. This opens the possibility of more compact detectors and maybe to relate with a better accuracy the neutron direction to the impinging antineutrino direction [9]. Unfortunately liquid scintillators doped with $^6$Li are not yet stable enough and R&D is on-going to develop new liquids [10]. The actual alternative is to use solid plastic detectors using $^6$Li:ZnS layers as we will see in the next section.

In order to comprehend the interests of the various detection media, one has to understand the possible background signals that could blur the antineutrino one. Cosmic rays constitute a recurrent background source, independently of the type of reactor to monitor. Usually fundamental neutrino physics experiments eliminate an important part of this background with underground deployments,

allowing large overburdens. Even at shallow depth the hadronic part of the background is eliminated, while remain cosmic muons mainly. These cosmic muons are at the origin of Michel electrons, fast neutrons and cosmogenic nuclei i.e. radioactive nuclei, both created through electromagnetic spallation or muon capture processes in the matter. Let's classify the backgrounds in two categories: the accidental background which mimics the antineutrino signal through random coincidences between for instance gamma rays from natural radioactivity of the rock or materials and fast neutrons arising from the cosmic muons ; and the correlated background which mimics the antineutrino signature with a prompt and a delayed signal arising from the same physics event. Fast neutrons can constitute a correlated background if they are moderated in the detection medium on protons which recoil limit the positron energy loss before being captured on the chosen absorbant (Gd or else). Cosmogenic isotopes which are beta–delayed neutron emitters can also be a source of correlated background. With above ground deployments, one has in addition to cope with the hadronic part of the cosmic rays i.e. mainly neutrons.

Different methods can be developed to help distinguishing between the background and antineutrino events. Antineutrino detectors should use low radioactivity materials to minimize accidental background. Lead shielding is also used around the antineutrino target to decrease the gamma ray background, which can be very strong at very short distance of a research reactor. Borated polyethylene shielding is used to decrease the neutron background reaching the target. Then usually the target detectors are complemented with an active cosmic muon veto detector, allowing to tag a muon interaction and preventing to measure events in a definite time window following the muon signal in the veto. In addition Pulse Shape Discrimination technique is used, especially with liquid scintillator targets, with the goal to distinguish neutron from gamma signals. Different target media can also be used to eliminate some background components.
The actual main techniques use these different methods at a different level: liquid scintillator based detectors, solid plastic based detectors and water based detectors [11].

## 2.1. Liquid scintillator based detectors

The liquid scintillator technique doped with Gd is the most mature detection technique employed for several decades. The properties observed in large detectors designed for fundamental physics are nevertheless hard to transpose to small scale detectors as edge effects combined with a limited number of PMTs affect the energy and spatial resolution one can obtain. The first demonstration of antineutrino detector for reactor monitoring was made by the SONGS experiment [12] with a very simple detector of modest detection efficiency that took data unattended for about one year. Since then, efforts have been concentrated towards an optimization of the detector performance with different designs. According to the simulation work developed to design small scale detectors, the energy resolution can be very good and the detection efficiency as well, depending on the size and number of PMTs used. In order to optimize the background rejection, the use of PSD is mandatory, especially for above ground deployments. Up to now high performance detectors still require shieldings and a cosmic muon veto around the target which increase a lot the detector footprint. There are several on-going efforts worldwide with different goals ; SONGS2 (3.6t detector) at a CANDU reactor in Point Lepreau Canada [11,13], the KASKA prototype first installed at the JOYO fast reactor and now installed above-ground at a PWR (with an on-going PSD R&D) [14], the Nucifer detector at the OSIRIS research reactor in France [15] (with an on-going PSD R&D and the goal of an optimized efficiency).

## 2.2. Solid plastic based detectors

An alternative to liquid scintillator target detectors is the use of solid scintillators. The use of solid scintillators prevents the safety problems that could arise from the use of liquids of rather low flash points. Another advantage is the possible segmentation of the detector. The segmentation ensures naturally a very good spatial resolution and allows further background elimination thanks to multiplicity cuts. A set of projects develops segmented plastic concepts with Gd layers on slabs, either read by PMTs or MPPCs with fibers: the DANSS (Russia) [11,13], CORMORAD (Italy) [16] and the PANDA (Japan) [11,13] experiments. All three experiments have deployed prototypes at reactors. In these designs, the searched prompt and delayed signals are still produced by photons (use of Gd sheets) and distinguished with an energy cut helped by the topology of the events. If we refer to the DANSS collaboration predictions for their design [11,13], the energy resolution is not as good as in liquid scintillator designs, but the spatial resolution is much better and the detection efficiency could be quite

high. No full scale detector has taken data yet, so these figures should be confronted to experimental measurements before drawing any conclusions. The full scale DANSS detector is under construction and should bring new results in the very next years. Results will be presented at the ESARDA NA/NT meeting in Bruges [11].

Another possibility is the use of solid segmented plastic detectors with $^{6}$Li:ZnS layers instead of Gd sheets. The use of $^{6}$Li allows the identification of the neutron through PSD information and the segmentation gives a precise location of the interaction. The neutron is captured after a lower energy loss in $^{6}$Li than in Gd, its range is thus shorter. This opens a potentiality for more compact detectors, and direction sensitive measurements.
The Sandia lab. in collaboration with LLNL have already tested a 4-cell prototype with organic scintillator and ZnS:Ag/$^{6}$LiF screens on outer surface at the San Onofre power station in 2011. The ability of such a design to reduce background thanks to the topology of the events was demonstrated. In Europe, the SOLiD collaboration has developed a different design using fibers for the light collection and MPPCs with electronics inherited from T2K [15,17]. In both cases a full scale detector should be built in the coming years allowing to appreciate the real quantitative improvements of this approach.

### 2.3. Water based detectors

Another way to eliminate background is to choose another detection medium, like water detectors in which $GdCl_3$ is dissolved. In these detectors, the detection reaction is still the IBD process. The prompt signal is provided by the Cerenkov light of the created positron in the water, and the delayed signal by the photons coming from the deexcitation of the Gd isotope after neutron capture. Only the particles generating a Cerenkov signal can be detected. This induces a quite high energy threshold on the reactor antineutrino detection [18] but also allows to eliminate a large part of the fast neutron background as neutrons of energy lower than 500MeV cannot be detected (because of the recoil proton Cerenkov threshold).
R&D is on-going as there are plans to fill in large detectors for fundamental neutrino physics [18]. Two initiatives of such detectors developed for safeguard purposes exist ; the Brazilian project ANGRA and the water detector deployed by LLNL at the San Onofre power station.
The construction of the ANGRA detector is nearly finished and the detector should be deployed above ground before the end of this year [11,13]. The detector of the LLNL is deployed above ground and the data taken are under analysis. The drawback of the water detectors is the impossibility to identify particles and a poor energy resolution. But these detectors are inherently safe thanks to the use of water.

### 2.4. Comparative properties

One objective of the next antineutrino detection sub-WG group meeting in Bruges is to show the results of these various experiments but also to try to find common ways to quantify the detection properties. We need to agree on common definitions of quantities to qualify the detector performances. This is mandatory to allow comparisons helping leading the further required R&D. This is also mandatory to be able to use the detector performances in the study of diversion scenarios, coupling reactor simulations to detection scenarios. These scenarios can also help leading the R&D by characterizing what are the detection properties to be improved in order to meet the required sensitivity on the fuel composition of various reactor types.

## 3. Reactor simulation initiatives

Several reactor simulation developments for reactor monitoring with antineutrino detectors are on-going worldwide. The first initiative came from the Double Chooz collaboration with the development of a simulation tool, the MCNP Utility for Reactor Evolution (MURE) code [19] and first scenarios involving PWR and CANDU reactors [20]. Since then several groups study diversion scenarios taking into account reactor physics constraints. The latter point is mandatory because these reactor physics constraints may eliminate de facto some scenarios that would be impossible to realize for safety reasons inherent to the operation of a reactor core. These reactor physics constraints influence also the results of a given scenario as the fission rate variations due to a fissile material diversion may be restricted by constraints on the maximal variation of the multiplication coefficient of the core or the fuel composition may be constrained by the minimum required delayed neutron fraction…

To study whether an antineutrino detector of a footprint of a cubic-meter would reach an accuracy sufficient to detect a "significant" diversion in a "timely" fashion, the studied scenarios usually adopt the following IAEA definitions: a significant quantity would be of 8kg of plutonium to be detected in the timeliness of 3 months.
IAEA is interested in knowing the response of an antineutrino detector associated to a lot of reactor designs and cases. Among the main ones, are the on-load reactors (CANDU but also Gen-IV Pebble Bed Reactors), the main Gen-IV designs, and the capability to distinguish various fuel compositions, like UOx vs MOX fuels or innovative fuels (including standard fuel burnt in non thermal reactors).
Several studies are on-going and will be presented at the ESARDA NA/NT WG meeting [21].
A common methodology animates the following studies: first checking the modelled reactor physics using most of the time benchmarks, then check the feasibility of the considered scenarios under reactor physics constraints and check the interest of the scenario for safeguards (quality, quantity of the diverted materials). The last step is to analyse the probability of detection of the diversion using statistical tests, provided a number of hypotheses on the detector location, size, detection efficiency...
A study is on-going at Georgia-Tech and LLNL to see if antineutrino detection could help monitoring the irradiation of plutonium-based 'MOX' fuel to ensure the material is hard to recover without reprocessing [22]. Two reactor designs are under study, using MCNPX and the CINDER evolution code: a Westinghouse-type PWR with partial MOX loading (most common on the US side) and the Fast reactor BN-600 with partial and full MOX loading (Russian side). The relation between the hypothetical detected antineutrino signal with the burnup is under study. Diversion scenarios of replacement of assemblies by LEU or dummy assemblies are also under study.
Recent results obtained with the MURE code were presented recently [23]. Scenarios for a PBR (Pebble Bed Reactor) and a sodium-cooled FBR (Fast Breeder Reactor) have been studied. These reactors are Gen IV reactors, presenting issues for safeguards purpose: a PBR is an onload refuelling reactor, easing the withdrawal of plutonium of good quality, while FBR can, by definition, build-up plutonium.

Sophisticated statistical methods inherited from fundamental neutrino physics calculations are also being used to compute the sensitivity of the antineutrino probe to the fuel content [24]. These calculations rely on assumptions made on the detector size, location, performance, but also on the reactor power. They were not yet coupled to realistic reactor simulations but one could imagine to do so in a near future as the statistical tests constitute the ultimate steps of the study of scenarios. One outcome of these studies is the limitation of the sensitivity of the antineutrino probe due to the actual uncertainties associated to the reactor antineutrino spectra [24]. This opens a natural link between our problematics and nuclear physics as the antineutrinos are emitted in the beta decays of the fission products.

## 4. Links with other physics fields

An important work has been performed recently to investigate the existing methods to compute the antineutrino energy spectra associated to the main isotopes contributing to the fissions in a Pressurized Water Reactor i.e. $^{235}$U, $^{239}$Pu, $^{238}$U and $^{241}$Pu. The method developed by Schreckenbach et al. to convert the reference ILL integral beta spectra [25] was revisited leading to a normalisation shift of the newly obtained spectra upward by 3 % [26,27]. In parallel, the method relying on the summation of all the beta decay branches of the fission products was revisited as well, taking benefit from the huge quantity of nuclear data available nowadays [26, 28], coupled to the MURE code. These works led to new synergies of our problematic with fundamental neutrino research on one hand with new experiments at reactors aiming at evidencing potential sterile neutrinos [29] and with nuclear physics measurements on the other hand to improve our knowledge of the reactor antineutrino spectra [28,30].

### 4.1 Nuclear Physics

The antineutrino spectrum associated with one of the 4 fissioning isotopes in a moderated reactor can be computed as the sum of the contributions of all fission products thanks to the use of the full information available per nucleus in nuclear databases. This so-called summation method is useful on several aspects. Not only it is the only one adapted to the computation of the antineutrino emission associated to various reactor designs, but also it allows the computation of antineutrino spectra for

which no beta spectrum was measured so far. Moreover it is one of the only alternatives to the ILL data whilst takes into account off-equilibrium effects and allows to work with different energy binning of interest for reactor neutrino experiment analyses.
Lately, new summation method calculations of the antineutrino energy spectra arising after the fissions of the four main fissible isotopes $^{235;238}$U, and $^{239;241}$Pu in PWRs were obtained. The new calculations include the recently measured beta decay properties of the $^{102;104;105;106;107}$Tc, $^{105}$Mo, and $^{101}$Nb nuclei, that were suspected to suffer from the Pandemonium effect [30]. These beta feeding probabilities, measured using the Total Absorption Technique (TAS) at the JYFL facility of the University of Jyvaskyla, have been found to play a major role in the gamma component of the decay heat for $^{239}$Pu in the 4-3000 s range [30]. Following the fission product summation method, the calculation was performed using the MURE evolution code coupled to the experimental spectra built from beta decay properties of the fission products taken in evaluated databases. These latest TAS data are found to have a significant effect on the Pu isotope energy spectra and on the energy spectrum of $^{238}$U. It has thus been shown that the Pandemonium effect plays a major role in the estimate of the antineutrino spectra [28]. TAS measurements can allow to improve drastically the prediction ability of these spectra. Moreover, independent evaluations of the reactor spectra could provide new constraints on the potential existence of sterile neutrinos but require a complete error calculation associated to the summation method spectra.

## 4.2 Neutrino Physics

As quoted above, the recent re-computation of the reactor antineutrino spectra has motivated a new search for sterile neutrinos at reactors with short baseline experiments [ref][ref]. Most of the projects quoted above are mainly motivated by safeguards, but have added a fundamental physics part to their physics case. Conversely, new projects aiming at measuring sterile neutrinos at reactors could bring new detection and sensitivity information valuable for non-proliferation. In addition to the projects already quoted above, one can quote the following projects: SCRAAM (LLNL US), Neutrino4 (Russia), POSEIDON (Russia), HANARO (Korea), and STEREO (France) [29].
A review will be presented at the ESARDA NA/NT meeting, with emphasy on the possible synergy with reactor monitoring using antineutrino detection [21].
In addition to these projects, large detectors deployed at power plants measuring the theta13 mixing angle, Double Chooz, Daya Bay and Reno, can bring new results regarding the antineutrino probe. First the R&D performed in the frame of these fundamental physics experiments triggered some of the R&D made for neutrino applied physics.
Secondly, their near detector measurements will provide sensitivity limits for probing the fuel content of a power reactor core when combined with the associated reactor simulation and operation parameters. In this frame, a detailed full-core simulation of the Chooz PWRs, the most common reactor design in the world, has been successfully developed last year with the MURE code in order to compute their antineutrino emission for the Double Chooz experiment [ref]. These simulations will be compared during the second phase of the Double Chooz experiment with the near-detector data, giving a limit to the sensitivity on the fuel composition and reactor power that antineutrino detection could bring. A careful estimate of the fission rate systematics has been done and could allow to maintain the error associated to the antineutrino prediction as low as 1.7% [31]. This simulation is the first realistic simulation of a reactor core performed in the frame of a reactor antineutrino experiment. This constitutes a key step of the studies of the detection of antineutrinos for non proliferation purpose as it will enable us to evaluate the sensitivity of this probe thanks to its accuracy. This validation is all the more important as we are moving toward an integrated tool (detector + simulation) for our non proliferation effort. Thirdly, the near detector measurements may constrain the value of the weak magnetism term entering in the antineutrino spectrum calculation. This term is subject to large uncertainties akin to affect the normalisation of the antineutrino spectra [32].

## 4.3 Neutron Detection

As we have seen in the first sections of these proceedings, the detection of a reactor antineutrino is signed by a delayed coincidence between the signals created by a positron and a neutron. There is thus obviously strong synergies between antineutrino detector and neutron detector R&D. For instance some segmented plastic detector designs quoted above have led to detectors that could possibly replace 3He counters. The STUK develops a design of neutron detector combining plastic scintillator and 10B. We thus have to create bridges between the two research communities which should allow to improve neutron and antineutrino detection efficiencies and share ideas [21].

## 5. Conclusions and outlooks

In these proceedings, we have attempted to review briefly the main experimental projects on-going for near-field reactor monitoring. Several complementary techniques are employed to reject backgrounds while preserving the antineutrino detection efficiency, either based on different detection media either based on geometrical properties. This large variety of designs is accompanied by a large variety of deployment sites and a lot of results can be expected in the coming 2 years.
All these research axes are the object of discussions and exchanges between nuclear facility operators, safeguard authorities and researchers in the frame of the sub-working group (WG) devoted to the antineutrino probe of the European Safeguards R&D Association (ESARDA [1]). This sub-WG is part of the NA/NT WG created in 2010 by the ESARDA. The goal of the sub-WG is to establish a road-map for the development and performances of antineutrino detectors for reactor monitoring. Regular meetings are organized and the next one will be held at the occasion of the ESARDA annual meeting in Bruges (Belgium) on May 27. 2013 [1] and is the object of these proceedings.
We expect very interesting discussions at this occasion, presenting the proceedings of the first Ad-Hoc WG meeting at IAEA [ref], review talks on the on-going studies, but also on the synergies with other physics areas… This material should help the community to draw the roadmap bringing the antineutrino detection on the next level of the Technology Readiness scale [ref].

## 6. Acknowledgements

The author would like to acknowledge Harri Toivonen and Julian Whichello for their support, which makes possible the organization of the next meeting of the antineutrino sub-WG in the frame of the ESARDA NA/NT WG. The author would like to thank in addition all the speakers and all the participants for their participation.

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# Proliferation resistance features of reprocessed uranium

K. Abbas, G.G.M. Cojazzi, G. Mercurio, P. Peerani, G. Renda
European Commission, Joint Research Centre - Institute for Transuranium Elements
Unit E08 - Nuclear Security - T.P. 800
Via E. Fermi, 2749
21027 Ispra, Italy

kamel.abbas@jrc.ec.europa.eu

***Abstract***

The introduction of reprocessed uranium in fresh LWR fuel as a measure against proliferation has been recently put forwards in the non-proliferation discussion. In particular the use of reprocessed uranium implies the presence of $^{232}U$ radioisotope and this can make more complicate the possible diversion of the fresh fuel as well as the further re-enrichment of the contained uranium.
This work attempts to further evaluate a composition of a proliferation resistant $UO_2$ fuel made of a mixture of fresh and reprocessed uranium implying i.a. the presence of $^{232}U$ and $^{236}U$ in the fresh fuel. Based on simulation a composition of new $UO_2$ fuel made half with fresh uranium and the other with reprocessed $UO_2$ is proposed. $^{232}U/^{208}Tl$ emits a high energy $\gamma$-ray of 2.6 MeV enhancing its detectability and any attempt for an undeclared enrichment to a weapon grade using for example centrifugation process could be complicated as the enrichment of $^{232}U$ will rise faster than that of $^{235}U$. Moreover the high energy $\alpha$ particles emitted by $^{232}U$ and $^{236}U$ might tend to dissociate $UF_6$ molecules, potentially complicating the centrifugation process. Regarding the radiation conditions of the proposed fuel, the calculations have shown that the dose rate at 1 m from a fuel assembly made of the proposed mixture remained in the same order of magnitude with respect to fuel made of non-reprocessed uranium although the dose rate doubled.

**Keywords:** $^{232}U$, $^{236}U$, proliferation resistance, safeguards, nuclear fuel cycle

## 1. Introduction

At present some countries are considering to introduce nuclear power while others are in process of expanding their current capacity. For these countries nuclear power allows them to move towards a secure zero-carbon emission electrical energy supply. Following the recent devastating accident at Fukushima most countries with running nuclear reactors are reviewing and reinforcing their safety and regulatory procedures rather than relinquishing this source of energy. For example in Japan, a political decision came to shut down all the nuclear power plants in the country however after a year and after some safety assessment, those nuclear power plants were re-switched on as there is no other possibility to satisfy the energy need of the country.

Presently there are about 500 commercial power reactors in operation or under construction worldwide and most of them require enriched uranium as fuel. The possible nuclear expansion raises concerns over the risk of diversion of nuclear materials through the expected growth in export or transfer of nuclear fuel. Moreover, recent revelations of possible clandestine nuclear programs have led to an increased global concern which has caused an increase in international efforts against proliferation of weapons of mass destruction. In this context new approaches are needed to prevent

such threats. One example is research and development effort on new proliferation resistant nuclear fuels.

According to IAEA [1]: “Proliferation Resistance is defined as that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by States intent on acquiring nuclear weapons or other nuclear explosive devices. The degree of proliferation resistance results from a combination of, inter alia, technical design features, operational modalities, institutional arrangements and safeguards measures.” These can be classified as intrinsic features and extrinsic measures.

In this paper focus will be mostly on intrinsic features, in particular the paper attempts to evaluate a composition of a proliferation resistant $UO_2$ fuel, made of a mixture of fresh and reprocessed uranium. The concept of introduction of reprocessed uranium in fresh fuel is already reported by several authors [2 - 6], including its effects on proliferation resistance. In this paper we focused on $UO_2$ fuel for LWR reactors. Reprocessed uranium contains, in addition to $^{238}U$ and $^{235}U$, several other uranium isotopes of interest, in particular $^{232}U$ and $^{236}U$. In section 2 the main features of $^{232}U$ are summarised. In section 3 the concept of reutilisation of reprocessed uranium is presented including an example of proliferation resistant fuel with reprocessed uranium. Section 4 illustrates some of the advantages of the utilisation reprocessed uranium from the point of view of proliferation resistance and safeguards, mainly from its detectability point of view.

## 2. $^{232}U$ characteristics and production

$^{232}U$ is a pure $\alpha$ emitter of 69.85 y half-life with two main $\alpha$ emissions at 5.32 and 5.26 MeV with intensities of 69% and 31% respectively. Figure 1 shows the main $^{232}U$ decay pathways. The relevant pathway with regard to this study is the one leading to $^{208}Tl$ (3.65 min), which emits a high energy $\gamma$-ray of 2.6 MeV (branching ratio > 99%).

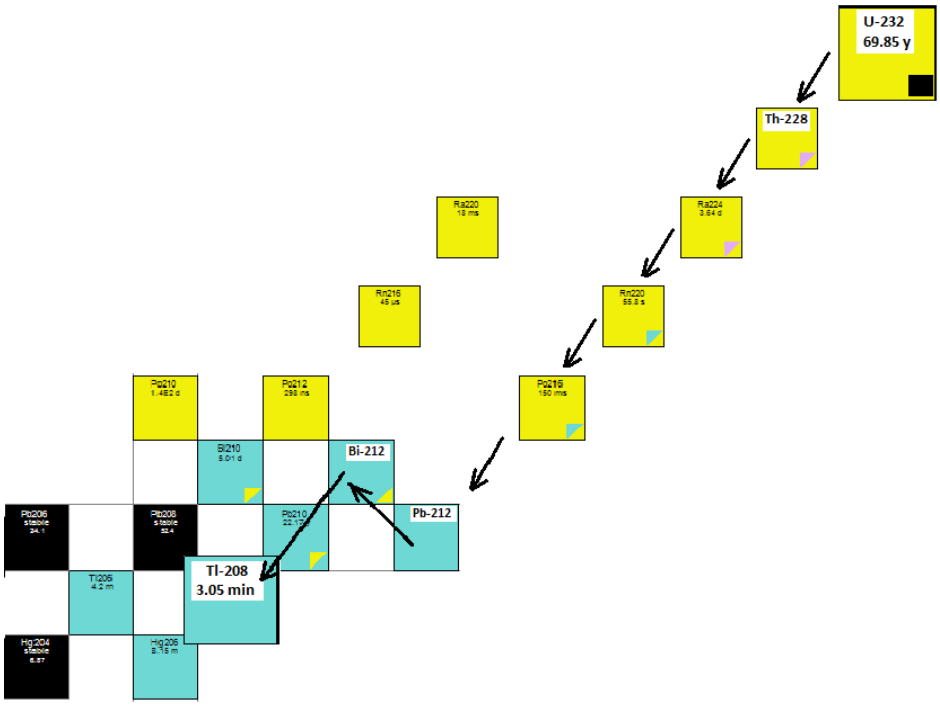


*Figure 1: Decay pathways of $^{232}U$.*

Figure 2 shows the build-up from $^{232}U$ of $^{208}Tl$ and other daughters such as $^{228}Th$ and $^{224}Ra$. Several years of $^{232}U$ decay are needed to build up $^{208}Tl$. This period could fit with a typical cooling time of spent fuel before reprocessing or disposal. Figure 3 presents measured and simulated $\gamma$-ray spectra of sample of 20 months old 3.7 MBq $^{232}U$ sample. The measured $\gamma$-ray spectrum which is measured in this work was acquired with a calibrated High Purity Germanium Detector (HPGe) of 50 % relative efficiency while the simulated one was calculated using Nucleonica program [7]. Qualitatively the 2 $\gamma$-spectra are in good agreement which is a proof on the reliability of Nucleonica program which is used in different steps of the present work. In both spectra, the $\gamma$-ray peak at 2.6 MeV is well resolved.

There are several production ways of production of $^{232}U$ in the nuclear fuel cycle (uranium or thorium fuels) via processes such as:

$^{237}Np(n,2n)^{236}Np - \beta - ^{236}Pu - \alpha - ^{232}U$
$^{234}U(n,2n)^{233}U - ^{233}U(n,2n)^{232}U$
$^{232}Th(n,2n)^{231}Th - \beta - ^{231}Pa(n,\gamma)^{232}Pa - \beta - ^{232}U$
$^{230}Th(n,\gamma)^{231}Th - \beta - ^{231}Pa(n,\gamma)^{232}Pa - \beta - ^{232}U$
$^{235}U - \alpha - ^{231}Th - \beta - ^{231}Pa(n,\gamma)^{232}Pa - \beta - ^{232}U$
$^{236}Pu - \alpha - ^{232}U$

The $\alpha$ decay of $^{236}Pu$ indicates that the production of $^{232}U$ rises with the cooling time of the irradiated fuel. $^{236}Pu$ is produced in small quantities from decay of $^{236}Np$ which is activated from (n,2n) and ($\gamma$,n) reactions on $^{237}Np$ during the reactor irradiation cycle.

As an indication of the uranium composition in a typical PWR spent fuel, the depletion code Origen contained in Scale program package [8] was used. Simulations performed in this work indicate that typically irradiated fuel ($UO_2$, 3-4% initial enrichment, burnup 40 GWd/t, 3 y irradiation) after a standard cooling time (~ 5 y) would contain, in addition to $^{238}U$ and $^{235}U$, several other uranium isotopes of interest. These include $^{232}U$, $^{233}U$, $^{234}U$ and $^{236}U$ of approximate concentrations of 1.0 $10^{-7}$, 3.2 $10^{-7}$,1.8 $10^{-2}$ and 5.0 $10^{-1}$ % respectively of total mass of uranium. The high $\gamma$-ray energy of 2.6 MeV emitted by $^{208}Tl$ daughter makes $^{232}U$ easily detectable and difficult to shield. Table 1 shows the results of Origen calculations simulating the uranium isotopic composition of 1 ton of $UO_2$ fuel enriched at 3.5%, irradiated at 40 GWd/tU during 3 cycles of 1 year each then left for 5 years decay. $^{208}Tl$ radioisotope content is also included.

*Table 1: Uranium isotopic composition in 1 ton of $UO_2$ fuel enriched at 3.5%, irradiated at 40 GWd/tU during 3 cycles of 1 year each and left for 5 years decay. $^{208}Tl$ radioisotope concentration is also included.*

| Uranium radioisotopes | Mass (g) | Concentration (mass %) |
|---|---|---|
| $^{232}U$ | $9.76\ 10^{-4}$ | $1.03\ 10^{-7}$ |
| $^{233}U$ | $3.04\ 10^{-3}$ | $3.21\ 10^{-7}$ |
| $^{234}U$ | $1.74\ 10^{2}$ | $1.84\ 10^{-2}$ |
| $^{235}U$ | $7.35\ 10^{3}$ | $7.76\ 10^{-1}$ |
| $^{236}U$ | $4.71\ 10^{3}$ | $4.97\ 10^{-1}$ |
| $^{238}U$ | $9.35\ 10^{5}$ | 98.70 |
| $^{208}Tl$ | $1.91\ 10^{-11}$ | $2\ 10^{-15}$ |

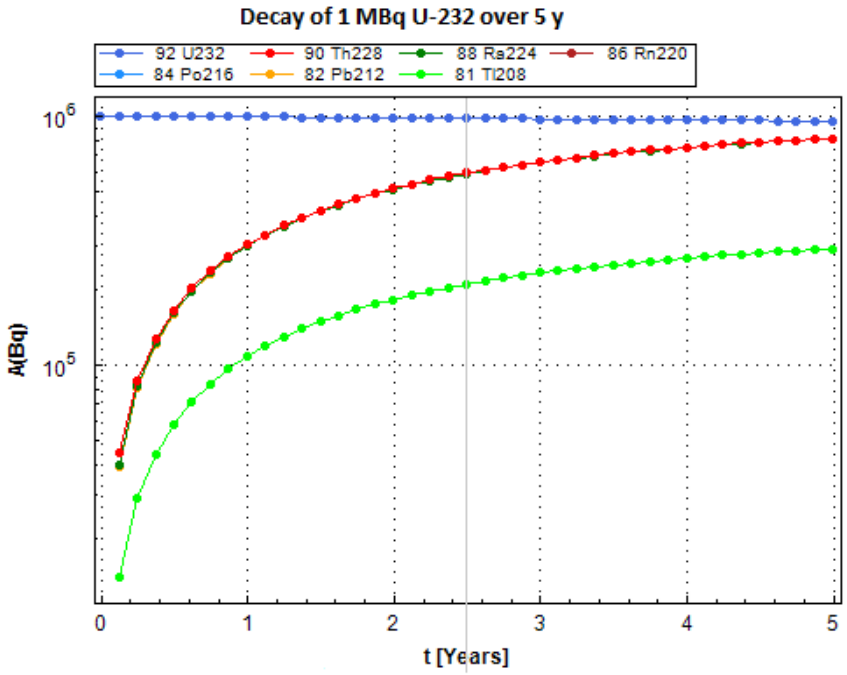


*Figure 2: $^{232}$U decay and build-up of $^{208}$Tl and other daughters calculated with Nucleonica program [8].*

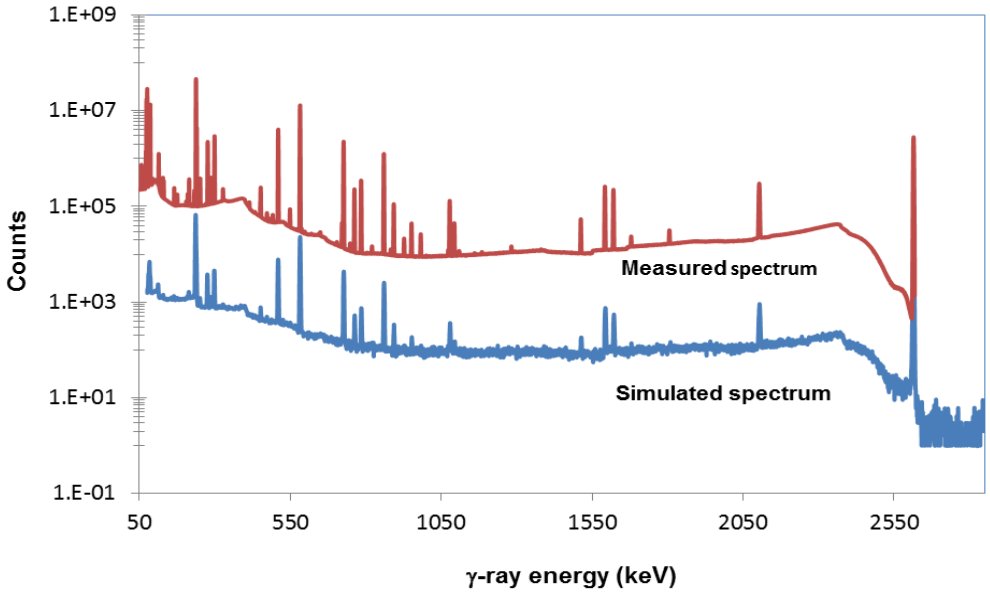


*Figure 3: Measured (top spectrum) and simulated (bottom spectrum) γ-ray spectra of a sample of 20 months old 3.7 MBq $^{232}$U. The measured γ-ray spectrum was acquired with a HPGe detector of 50 % relative efficiency.*

## 3. Re-utilisation of reprocessed uranium for fabrication of proliferating resistant nuclear fuel.

As mentioned above, the possible expansion of the nuclear energy will face new challenges to ensure a safe and sustainable use of this source of energy. The LWR is a proven technology and many

countries are building or planning to build such reactor types such as in China and in other emerging countries. At first stages of development, it is expected that an increased movements of nuclear fuel would occur all over the world. Consequently, the threat of nuclear material diversion would rise if new safeguards and security measures are not undertaken.

From the nuclear fuel cycle standpoint, the fabrication of the fuel and the uranium enrichment are sensitive technologies that need more and more tight controls. According to some investigators even the possibility that a sub-national terrorist group could build an enrichment facility should be considered [9]. In this case, low enriched uranium (LEU, enrichment < 20%) could be stolen from export deliveries for further enrichment needed for weaponisation intentions. Furthermore, it is worth mentioning that enrichment from LEU level to HEU is much less energy consuming with regard to the enrichment from natural uranium to the LEU levels.

The reprocessed uranium contains $^{232}$U and $^{236}$U radioisotopes which deserve some attention from the point of view of its re-utilisation for manufacturing of proliferation resistant fuel. $^{232}$U and $^{236}$U present safety issues for the nuclear industry if their concentration leads to dose limits exceeding e.g. those of fresh MOX fuel [4]. It is worth mentioning that as these 2 uranium isotopes are produced during the fuel irradiation, their analysis in spent fuel is useful as it can provide forensic information on the irradiation conditions and history.

The presence of large amount of $^{232}$U in uranium might necessitate handling of nuclear fuel under heavily shielded hot cells due to the worsen radiation dose emitted by the high $\gamma$-ray energy of $^{208}$Tl daughter. In this case, remote handling of such material would be necessary to prevent exceeding radiation rate exposure to the operating personnel. The content of $^{236}$U at high amount in the reactor core fuel can be also an issue for the operation of the reactor since this isotope is a high consumer of neutrons in a reactor core meaning that its concentration needs to be controlled and any excess neutron absorption has to be addressed and compensated to ensure a nominal $^{235}$U fission rate.

This paper proposes to manufacture new nuclear fuel made of fresh uranium mixed with reprocessed one that contains $^{232}$U which constitutes an intrinsic barrier against nuclear proliferation [9]. The proposed composition of new $UO_2$ fuel is made half with fresh $UO_2$ and the other with regenerated $UO_2$. $^{232}$U and $^{236}$U concentrations in the resulting fuel are kept in the same order of magnitude of those in a typical irradiated $UO_2$ nuclear fuel while the enrichment of $^{235}$U is adjusted to a common enrichment level of LWR fuel (3 – 4 %). An example of the uranium composition in a proposed $UO_2$ fuel is presented. Based on a scenario simulated with Origen code, this fuel is obtained by mixing the reprocessed uranium regenerated from 1 ton irradiated $UO_2$ (40 GWd/tU, 3 years irradiation, 5 years cooling) with 1 ton fresh 6% $UO_2$ which makes an overall enrichment of the new fuel of 3.46 %. Also based on Origen calculations, the proposed fuel if irradiated in the same conditions and left for 5 years cooling would have uranium composition as presented in the Table 2. As already mentioned above, the concentrations of $^{232}$U ($5\ 10^{-8}$ and $2\ 10^{-7}$%) and $^{236}$U (0.3 and 0.7%) in the proposed fresh and irradiated fuel remain in the same order of magnitude with respect to those of irradiated fuel made of fresh uranium ($^{232}$U: $1\ 10^{-7}$% and $^{236}$U: 0.5%).

Figures 4, 5 and 6 present simulated $\gamma$-ray spectra of respectively reprocessed uranium from spent $UO_2$ fuel (no regenerated uranium), uranium in the fresh proposed $UO_2$ fresh fuel (3.46% enriched, made of mixture of fresh $UO_2$ (6% enriched) and reprocessed uranium) and reprocessed uranium from irradiated proposed $UO_2$ fuel (40 GWd/tU, 3 years irradiation and 5 years cooling). In the three $\gamma$-ray spectra, the $\gamma$-ray peaks of $^{208}$Tl are well resolved which constitutes a precise signature of the fuel at high $\gamma$–ray energy.

*Table 2: Uranium isotopic composition in an example of new 3.46% enriched $UO_2$ fuel. The proposed fuel is composed of reprocessed uranium (regenerated from 1 t $UO_2$ irradiated fuel) mixed with 1 t of fresh 6% enriched $UO_2$. The irradiations conditions refer to 40 GWd/tU, 3 years irradiation followed by 5 years cooling.*

| Uranium radioisotopes | Concentration in spent $UO_2$ fuel (mass %) | Concentration in the proposed 3.5% enrichment $UO_2$ fuel (mass %) | Concentration in the irradiated proposed 3.5% enrichment $UO_2$ fuel (mass %) |
|---|---|---|---|
| $^{232}U$ | $1.03\ 10^{-7}$ | $5.01\ 10^{-8}$ | $2.16\ 10^{-7}$ |
| $^{233}U$ | $3.21\ 10^{-7}$ | $1.56\ 10^{-7}$ | $4.45\ 10^{-7}$ |
| $^{234}U$ | $1.84\ 10^{-2}$ | $3.64\ 10^{-2}$ | $2.04\ 10^{-2}$ |
| $^{235}U$ | $7.76\ 10^{-1}$ | 3.46 | $5.89\ 10^{-1}$ |
| $^{236}U$ | $4.97\ 10^{-1}$ | $2.56\ 10^{-1}$ | $7.08\ 10^{-1}$ |
| $^{238}U$ | 98.7 | 96.2 | 98.7 |
| $^{208}Tl$ | $2\ 10^{-15}$ | $9.83\ 10^{-16}$ | $4.28\ 10^{-15}$ |

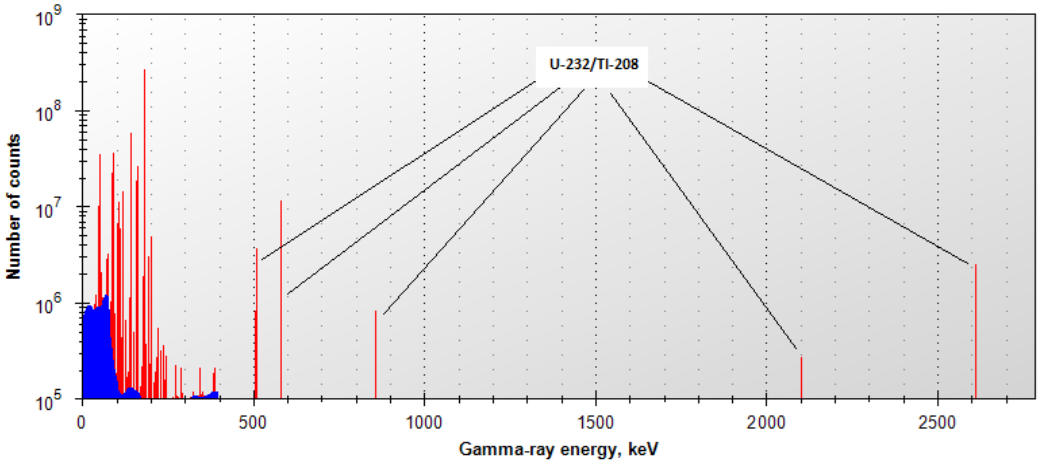


*Figure 4: Nucleonica simulated $\gamma$-ray spectrum of reprocessed uranium from spent $UO_2$ (3.5% enriched) irradiated during 3 years followed with 5 years cooling. Only contributions of uranium nuclides and $^{208}Tl$ are simulated for clarity.*

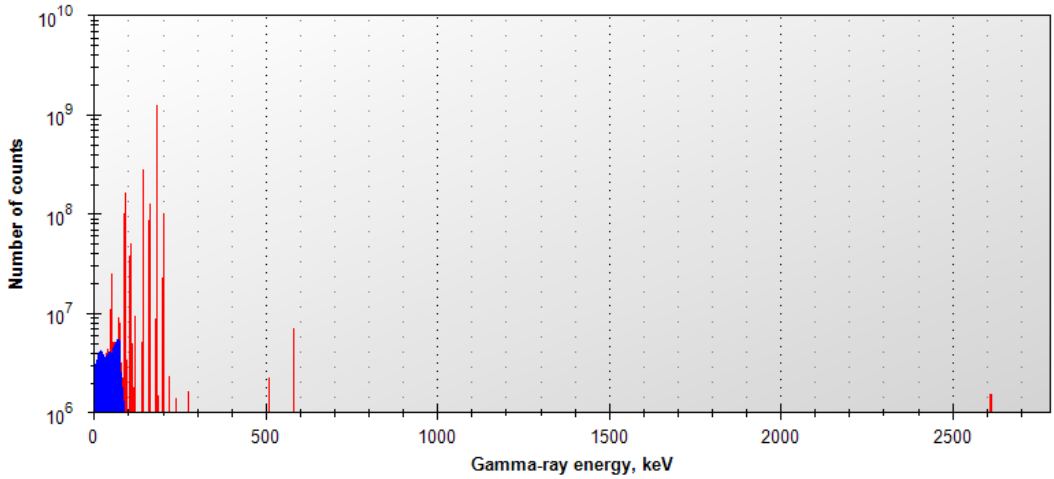


*Figure 5: Nucleonica simulated γ-ray spectrum of a proposed fresh $UO_2$ fuel (3.46% enriched). This fuel is made of reprocessed fuel mixed with fresh 6% enriched $UO_2$ fuel. Only contributions of uranium radioisotopes and $^{208}Tl$ are simulated for clarity.*

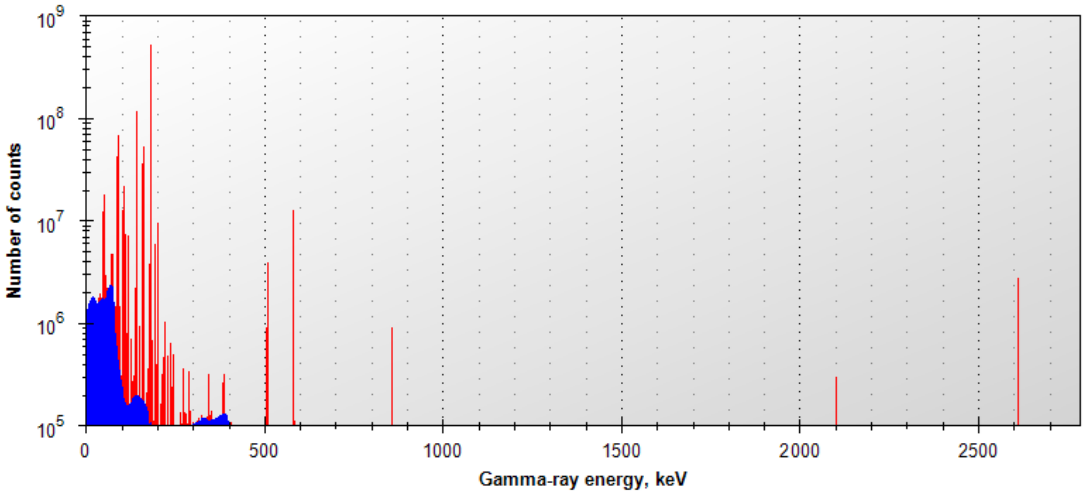


*Figure 6: Nucleonica simulated γ-ray spectrum of reprocessed uranium from irradiated fuel as in figure 5. Only contributions of uranium radioisotopes and $^{208}Tl$ are simulated for clarity.*

## 4. Some advantages of the utilization of reprocessed uranium for new $UO_2$ proliferation resistant fuel

The proposed nuclear fuel is based on the use of the reprocessed uranium which can present a significant savings of natural uranium resources. The re-utilisation of generated uranium in the conception of new fuel more proliferating resistant as proposed in this work would also contribute to the diminution of the stock of high enriched uranium, where available, which is another aspect in the fight against nuclear proliferation [6].

The presence of $^{232}U$ at a moderated concentration in the proposed fuel is an advantage as the detectability of the fuel is enhanced due the high penetrating γ-ray emitted by the $^{208}Tl$ daughter. From the point of view of neutron balance in the nuclear reactor, the impact of $^{232}U$ is negligible due to the low content of this radioisotope. However, the presence of $^{236}U$ affects the neutronics in the reactor if

the concentration is high which would necessitate shortening of the irradiation cycles. In the case of VVER-1000 fuel, Smirnov et al. reported that the $^{235}$U enrichment should be increased to overcome the issue of the $^{236}$U content [4].

The high energy γ-ray constitutes an advantage in terms of detectability in the fuel in the sense that presence of $^{232}$U/$^{208}$Tl would require considerable shielding for hiding it, therefore this would discourage any attempt of its diversion for use in an undeclared activity, including attempt of smuggling the material. Figure 7 shows Nucleonica simulated γ-ray spectra of the proposed uranium fuel composition that illustrate the detectability of the material in the case of using for example 20 cm lead shielding. It is clearly shown that the detection of the proposed fuel remains possible even with that shielding (γ-ray peaks of $^{208}$Tl well resolved) while for fuel without $^{232}$U (no high γ-ray peaks) a drastic attenuation of γ-radiation below 1.5 MeV occurs making possible hiding of the material. Moreover, it has a disadvantage of making any operational manipulation of the material more complex due to higher radiation doses. This is to say that activities of such material should be carefully kept below certain limits to ensure the "as low as possible" dose exposure for the handling operators and for the environment [4, 5].

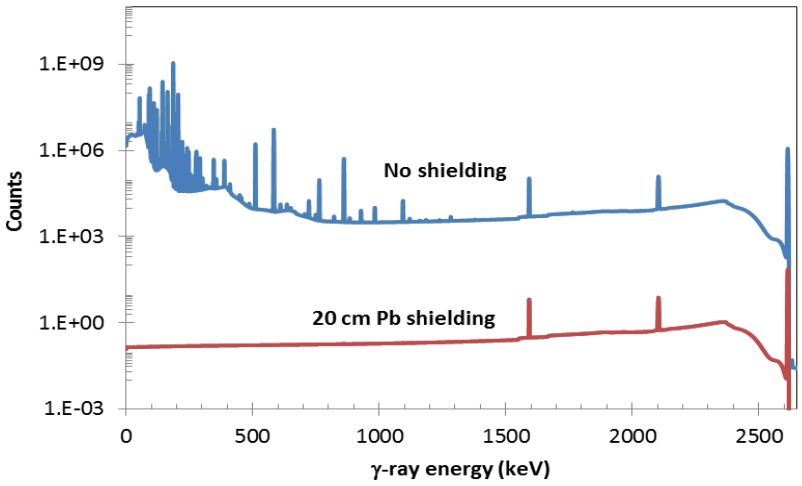


*Figure 7: Simulated γ-ray spectra of the proposed uranium fuel composition. The bottom and top spectra are respectively simulated with and without 20 cm lead shielding showing the drastic attenuation of γ-radiation below 1.5 MeV.*

The handling, transport and storage of fuel have to be addressed for any development of new nuclear fuel with respect to the assessment of the dose rate. In the case of the example of fuel composition treated above and based on Nucleonica simulation, the dose rate at 1 m of 500 kg (about 1 fuel assembly of PWR fuel) of 3.5% enriched uranium fuel is about half (~ 33 μSv/h) of that of the same assembly made of the fuel proposed in this work (3.46% enriched made of mixture of fresh and reprocessed uranium). Therefore the proposed fuel composition does not increase in a significant way the dose rate (to ~ 66 μSv/h at 1 m from a fresh assembly) and the radiation protection issues nor it will contribute to proliferation resistance in terms of radiation barrier on the other hand it can enhance proliferation resistance due to the easier detectability of the material. In particular the capability to remotely detect the material enables the implementation of techniques for item tracking and process monitoring, increasing the possibility to detect diversion of fuel from the facility.

It is worth mentioning that the presence of $^{232}$U and daughters in the proposed mixture does not mask the 186 keV γ peak from $^{235}$U. Indeed neither $^{232}$U nor its daughters produce significant γ-lines in the range of 150-220 keV, which is the range of interest for $^{235}$U identification, how it can be seen from Figure 8. In this figure we show simulated γ-spectra in high resolution germanium detectors, the absence of interfering lines between $^{232}$U and $^{235}$U would make still possible the use of low resolution NaI spectrometers, as frequently used nowadays for un-irradiated uranium.

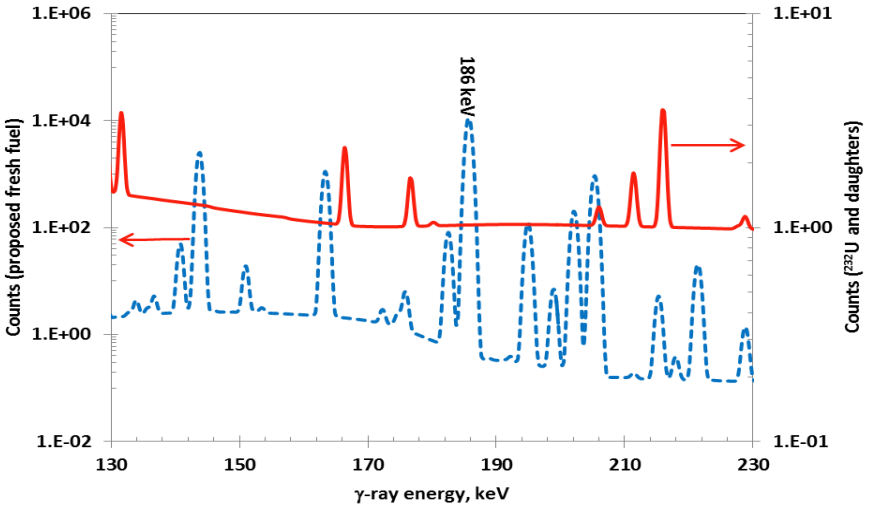


*Figure 8: Simulated γ-ray spectra in the energy range 130-230 keV of the proposed uranium fuel (dashed line) and of $^{232}$U including its daughters (continued line) showing no γ-peak interferences between $^{232}$U and $^{235}$U.*

In addition any attempt to re-enrich the proposed uranium to a weapon grade using for example centrifugation process would become problematic as the enrichment the $^{232}$U will raise faster than of $^{235}$U worsening so the radiation condition in an enrichment facility [4, 5]. Therefore the intrinsic radiation could reach worsen conditions that impact material handling.

Moreover in the context of 20 % enriched uranium doped with $^{232}$U, [9] shows that the α particles emitted by $^{232}$U and $^{236}$U could dissociate the $UF_6$ molecules inducing uncontrolled release of fluorine and making the enrichment to fail thus deters any potential proliferators. Given the large difference in the $^{232}$U concentration, the quantification of these effects for the mixture proposed in this paper is currently under estimation.

## 5. Conclusions

A concept of fabrication of proliferation resistant $UO_2$ fuel based of the utilisation of reprocessed uranium has been proposed in this work. Based on Origen code calculations, it is shown that for a mixture made of half of fresh uranium enriched at 6% and half of reprocessed uranium resulting in new (‘fresh”) $UO_2$ fuel enriched at about 3.46%, the concentrations of $^{232}$U and $^{236}$U would remain in the same order of magnitude as those of a spent fuel originally fully made of fresh uranium. Other similar and consistent scenarios could be obtained by mixing in different proportions the reprocessed uranium with uranium at any enrichment level.

Both $^{232}$U and $^{236}$U radioisotope presence in fuel were considered as complications in the nuclear fuel cycle as $^{232}$U rises the radiation field of the fresh fuel while $^{236}$U highly absorbs neutrons that need to be compensated by using higher enrichment. In this work, the proposed $UO_2$ fuel makes benefit of moderated concentrations of those uranium radioisotopes ($^{232}$U and $^{236}$U) to enhance non-proliferation and safeguardability of the nuclear material, in particular its detectability. In fact, fresh fuel containing $^{232}$U is difficult to hide as it emits γ radiation of high energy.

Moreover any attempt to enrich (with centrifugation) uranium from LEU to HEU for an undeclared activity would be made more difficult as the concentration of $^{232}$U will grow more than that of $^{235}$U. The quantification study of the dissociation of $UF_6$ due to the α particle is under investigation.

The authors are under way to investigate further the irradiated proposed fuel, fuel as mixtures of fresh and recycled uranium with plutonium and the use of multi-recycled uranium.

## Acknowledgements

The authors are grateful to Patricia Mortreau and Riccardo Rossa for the helpful discussions.

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# Promising NDA Technologies for Material Accountancy of Nuclear Material in Debris of Melted Fuel of Fukushima-Daiichi NPP

**Michio Seya,**
**Hideo Harada, Fumito Kitatani, Mitsuo Koizumi, Harufumi Tsuchiya Ryoichi Hajima, Takehito Hayakawa, Toshiyuki Shizuma, Christopher Angell, Alan Michael Bolind.**

Japan Atomic Energy Agency
Tokai-mura, Ibaraki-ken, Japan

***Abstract:***

*This paper describes some introduction of JAEA papers in the session "Post Fukushima : NDA Techniques for Molten Fuel Debris". We present at first conceptual methods for the removal of melted fuel (MF) in the reactor and containment vessels of Units 1 - 3 of Fukushima-Daiichi NPP. By categorizing removed MF, we could stratify debris of MF into particle-like debris and cut / small rock-like debris. Then we show two promising NDA technologies from the view point of nuclear material accountancy of melted fuel debris.*
*One of the two is Neutron Resonance Densitometry (NRD) based on NRTA (Neutron Resonance Transmission Analysis) and NRCA (Neutron Resonance Capture Analysis) utilizing the TOF (Time of Flight) method, which is intended for the analysis of particle-like debris. We JAEA and JRC-IRMM have already started collaboration on NRD since May 2012. This paper describes the proposed NRD very roughly, while the some of the papers show the details of this method.*
*The other NDA technology uses LCS (laser Compton scattered) gamma-rays (mono-energetic gamma-rays) to analyse nuclear material in cut / small rock-like debris. The application of the Nuclear Resonance Fluorescence (NRF) method for another type of debris using LCS gamma-rays is presented here very roughly. A paper in this series shows in detail the demonstration system of generation of intense mono-energetic gamma-rays using Energy Recovery Linac (ERL) and laser storing cavity. The demonstration system is just under construction in KEK Tukuba. The measurement techniques of scattered method and transmission method using NRF are also shown roughly. The last two 2 papers present these methods in detail.*

**Key words:** Debris of Melted Fuel, Fukushima-Daiichi NPP, Neutron Resonance Densitometry, laser Compton scattered gamma-rays, Nuclear Resonance Fluorescence

## 1. Introduction

The severe loss-of-coolant accidents at the Fukushima-Daiichi Nuclear Power Plant (1F-NPP) caused by the earthquake and the tsunami on March 11, produced melted fuel (MF) in the reactor cores of Units 1 – 3. Accident analysts are estimating that all of the core fuel, or a very large portion of it, melted into a combination of massive rocks and many small particles during these accidents. Nuclear reactors are usually classified as being "item facilities" in international safeguards; however, the severe accidents caused the items in these reactors (i.e., the fuel assemblies) to melt into rocks or particles (grains) and mix with structural materials. This means that Units 1-3 of 1F-NPP turned into bulk facilities, where nuclear material accountancy based on measurements is required. Quantifying the NM in MF debris by destructive assay (DA) is considered to be difficult because of the insolubility and unknown representativeness of the debris samples. Quantifying by conventional non-destructive assay (NDA) is also difficult because of the variety in the sizes and shapes of the debris particles and their unknown compositions. DA and NDA techniques for measuring the quantity of NM in MF formed in severe accidents have been studied for a very long time, since the TMI-2 (1979) accident and the Chernobyl-4 (1986) accident. However applicable measurement techniques have not been established so far.
It is currently being thought that the complete decommissioning of Units 1-4 of 1F-NPP will take 30 to 40 years. During this long-term decommissioning process, the removal of MF in Units 1-3 will most

probably start from 2021 or later and will continue for 10 to 15 years. Taking into consideration the available time prior to the removal of the MF from the reactors of Units 1-3, we have time to develop some techniques by improving conventional NDA technologies and creating cutting-edge technologies. Then we could apply the techniques to quantify the NM in the MF.
This paper presents the results of rough feasibility studies on two NDA technologies for the quantification of all relevant isotopes for accountancy of NM in MF generated by severe accidents, such as at 1F-NPP. One is NRD, which is based on semi-conventional NRTA and NRCA with TOF and which has the possibility of quantifying NM in the particle-like debris of MF [1]. Another is the NRF method, a cutting-edge technology that uses intense, high-energy, LCS, mono-energetic gamma-rays and has the possibility of directly measuring NM isotopes in small rock-like debris or cut pieces from large debris [2].
At first, we show the possible stratification of the MF debris according to the shapes of the particles, taking into account the methods of the removal of the MF from the reactor vessels or containment vessels.

## 2. Stratification of MF at the time of removal

At present, how we remove MF from the reactor and containment vessels of Units 1 - 3 of 1F- NPP is not under consideration, because there are many urgent things to do. However, for the development of techniques for the precise measurement of NM in the MF debris for material accountancy, we need to expect what types of MF (strata) are going to be handled during the removal in the future.
By making the removal methods clear, the removed MF can be categorized into several strata: cut debris (slab-type debris, cylindrical debris etc.), small rock-like debris and particle-like (grain) debris.
It is said that in the reactor vessels / containment vessels, MF are present in the forms of grains (particle-like debris) and large rocks. Here we show some expected methods for the removal of these MF from the reactor vessels / containment vessels.

### 2.1 Removal of large rock of solidified melted fuel

Figures 1 through 4 show four methods for removing the MF: breaking by a hard stick, drilling, cutting by a disk cutter, and cutting by a laser (or plasma).

### 2.2 Removal of particle–like solidified melted fuel

It is said that the rapid cooling of MF by water causes the MF to explode into many small grains of debris (particle-like debris). Also at the time of the removal of MF by the above methods, grains or small particle debris are generated. One method of taking grains out with water is shown in Figure 5.
We may summarize the strata of removed solidified melted fuel in Table 1, showing possible NDA techniques for quantifying the NM in it.

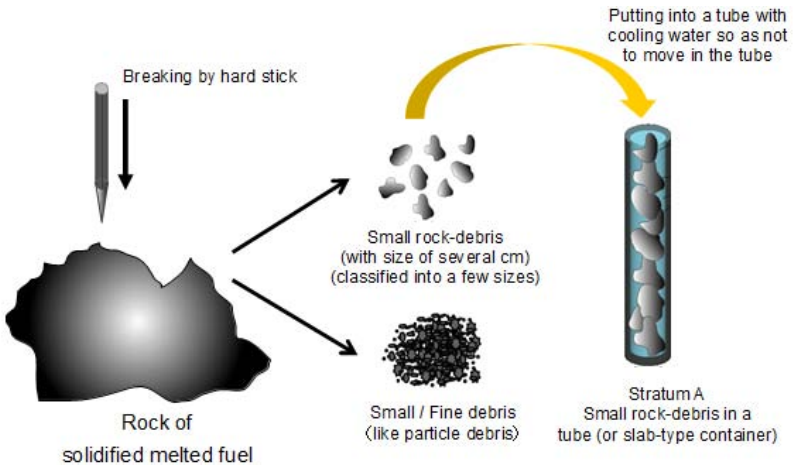


Figure 1 Removal Method 1 (Breaking into small debris with a hard stick)

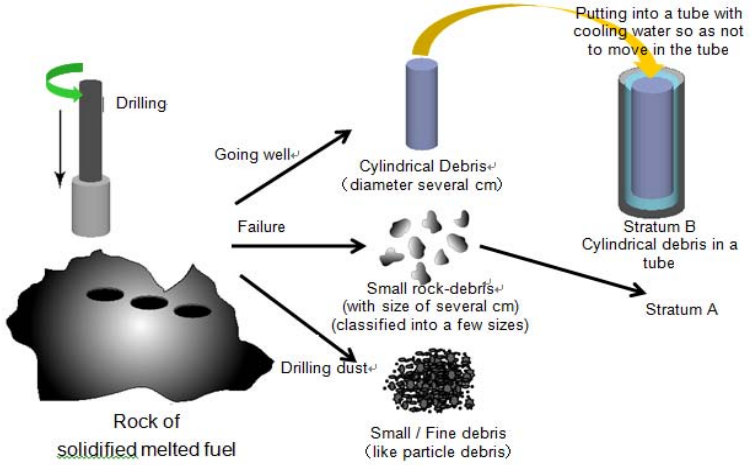


Figure 2 Removal Method 2 (Drilling (into cylindrical shape))

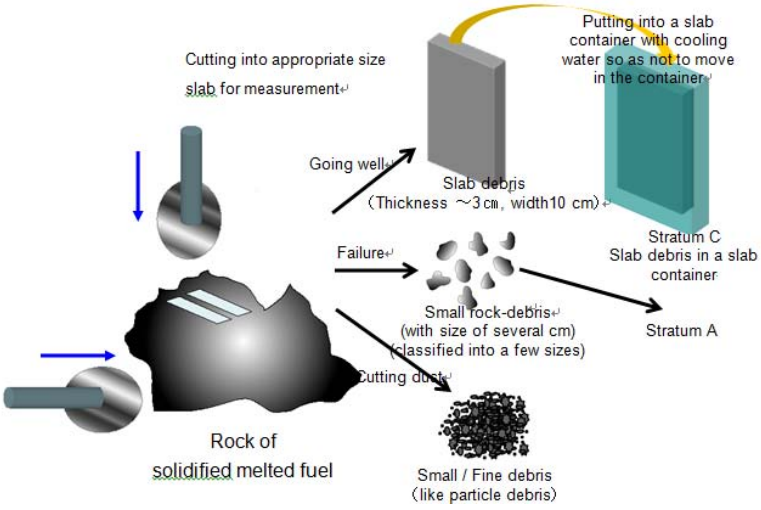


Figure 3 Removal Method 3 (Cutting by disk cutter)

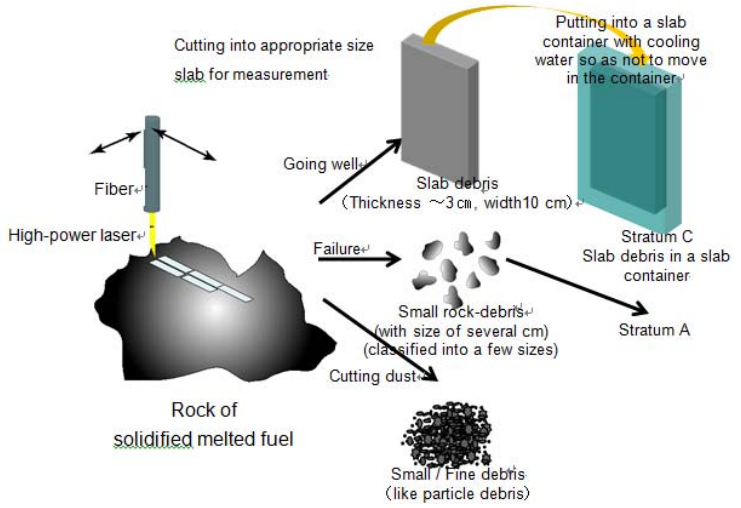


Figure 4 Removal Method 4 (Cutting by Laser (or Plasma))

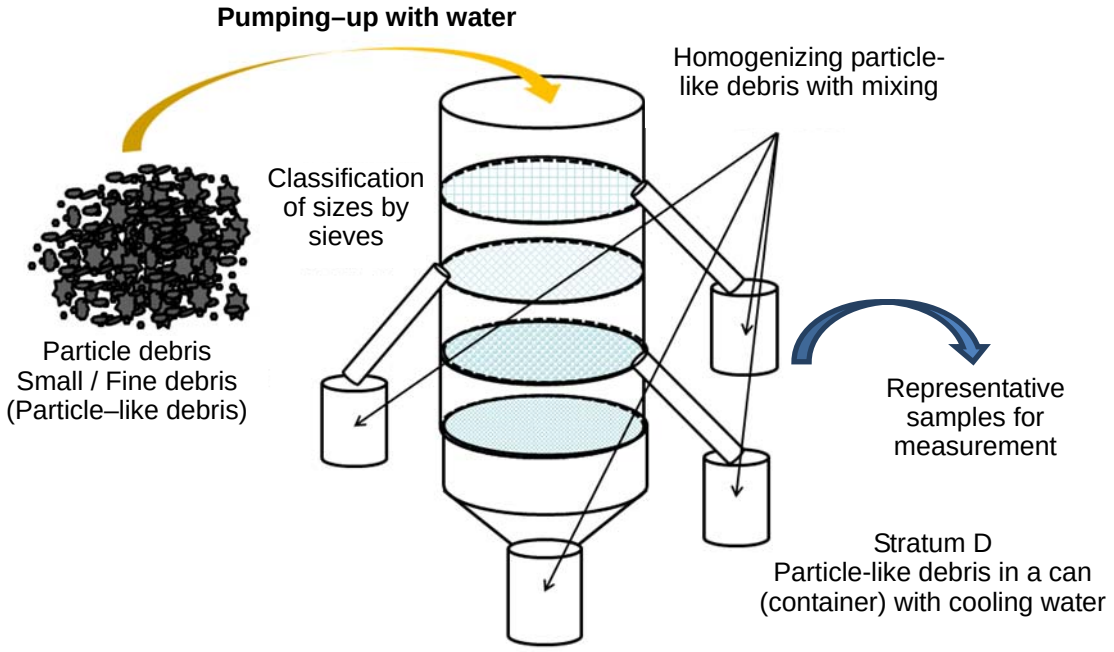


Figure 5 Removal of particle–like debris

Table 1 Strata of removed solidified melted fuel

| Stratum | | Explanation | Possible NDA |
|---|---|---|---|
| I<br>cut / small rock-like debris | A | Small rock-like debris (several centimetres in size and classified into several sizes), placed in a tube or slab-type container | -NRF using LCS γ-rays |
| | B | Cylindrical debris (several centimetres in diameter), placed in a tube | |
| | C | Slab debris (thickness ≈ 3cm, width ≈ 10 cm), placed in a slab container | |
| II<br>particle–like debris | D | Particle–like (grain) debris, placed in a thin disk container | -NRD (NRTA+NRCA) |

## 3. Rough Consideration of Feasible NDA Techniques for Precise Measurement

For seeking feasible NDA techniques for precisely quantifying all relevant isotopes of NM ($^{235}$U, $^{238}$U, $^{238}$Pu, $^{239}$Pu, $^{240}$Pu, $^{241}$Pu, $^{242}$Pu etc.) in MF debris by direct measurement, it is useful to think about the 14 techniques that are under investigation in the Next Generation Safeguards Initiative (NGSI) of U.S. Department of Energy (U.S. DOE) for the measurement of the mass of Pu in spent fuel assemblies [3]. These 14 NDA techniques are summarized in Table 2, being divided into neutron techniques and photon (X-ray or gamma–ray) techniques.

Among the techniques in Table 2, TN and PG quantify the burn-up of the spent fuel, from which we need to estimate the quantity of U/Pu based on calculations. Therefore, these two are not useful for the direct measurement of NM. In the application of these 14 techniques to fuel assemblies, the geometry of the sample (i.e., the fuel assembly) is well known. The composition is also roughly known. For example, it is safe to assume that the fuel is in an oxide form. This knowledge of the geometry and general composition is essential for many of these NDA techniques. This information is used to perform computer simulations of the expected behaviour of the instrument with various fuel assemblies. The experimental data from measurements on unknown fuel assemblies are interpreted by comparing them against these simulated data and/or calibration data.

Table 2 The 14 NDA techniques under investigation in NGSI
(*) P: Passive, A: Active (**) For NRF in NGSI, Bremsstrahlung gamma-rays was under consideration

| | P/A (*) | Name of NDA | What is quantified | Measurement of |
|---|---|---|---|---|
| Neutron | P | TN | Burnup | Spontaneous / Self-induced fission neutrons scattered many times in the objective (with support of MCNP simulation calculations for neutron behaviours) |
| | | PNAR | $^{239}$Pu-eff (PNAR) | |
| | | SINRD | $^{239}$Pu, $^{235}$U | |
| | | NM | $^{239}$Pu-eff (NM) | |
| | | DDSI | $^{239}$Pu-eff (DDSI) | |
| | A | CIPN | $^{239}$Pu-eff (CIPN) | Induced fission neutrons scattered many times in the objective (with support of MCNP simulation calculations for neutron behaviours) |
| | | DDA | $^{239}$Pu-eff (DDA) | |
| | | DN | $^{239}$Pu-eff (DN) | |
| | | LSDS | Total mass of $^{235}$U, $^{239}$Pu and $^{241}$Pu | Interrogated/transmitted neutrons through the objective |
| | | NRTA | All of U/Pu-isotopes | Interrogated/once transmitted neutrons through the objective |
| Photon | P | PG | Burnup | All passive gamma-ray lines from the objective |
| | | XRF | Relative mass of U to Pu | U and Pu X-rays (stimulated by the radiation emanating from the spent fuel) from the surface of the objective |
| | A | DG | Relative amount of $^{235}$U, $^{239}$Pu and $^{241}$Pu | Delayed gamma-rays emitted by fission fragments in the seconds following induced fissions |
| | | NRF(**) | All isotopes All of U/Pu-isotopes | Very high intensity mono-energetic gamma-rays interrogated /once scattering or transmitted gamma-rays through the objective |

| | | | |
|---|---|---|---|
| TN: | Total (gross) Neutron counting | PNAR: | Passive Neutron Albedo Reactivity |
| SINRD: | Self-Interrogation Neutron Resonance Densitometry | NM: | Neutron Multiplicity counting |
| DDSI: | Differential Die-away Self Interrogation | CIPN: | $^{252}$Cf Interrogation with Prompt Neutron detection |
| DDA: | Differential Die-Away analysis | DN: | Delayed Neutron detection |
| LSDS: | Lead Slowing Down Spectrometry | NRTA: | Neutron Resonance Transmission Analysis |
| PG: | Passive Gamma counting | XRF: | X-ray Fluorescence measurement |
| DG: | Delayed Gamma detection | NRF: | Nuclear Resonance Fluorescence measurement |

If the fuel composition is grossly different from what is expected, though, then the NDA signal will also be grossly different from what is expected. At that point, most of the NDA techniques cannot give precise information. The only option is to quarantine the fuel assembly until other, more precise analysis can be performed on it, such as destructive analysis.

In stark contrast to fuel assemblies, debris from melted fuel has random or ill-defined geometry and a very much unknown composition. The debris cannot be accurately simulated by Monte Carlo computer codes. Therefore, many of the NDA techniques are ineffective for measuring the fissile content of this debris; they can indicate only that the debris is not in the form of a standard fuel assembly—a fact which is already known.

The only way to solve this problem is to use NDA techniques that do not rely so heavily on the geometry and rough composition of the sample. By reducing the number of other unknown variables in the technique, we can compensate for these two unknowns. In most techniques, the origin, location,

energy, and time of the neutrons or photons are partly or completely unknown. Passive neutron techniques, in particular, know that the initial energy of the neutrons is fast; but the origins of the neutrons (for example, spontaneous fission, (α, n) reactions, or induced fission), the locations where they are produced, their energies during moderation, and the times at which they are produced are all unknown. Most active techniques know more information, such as the origin of most of the photons or neutrons (i.e., the very strong external source), the location of this external source, and—if the external source is pulsed—the starting time of these external photons or neutrons. However, if the photons or neutrons scatter many times in the sample before they are finally detected, then the locations, energies, and times of these particles become unknown again.

The ultimate solution should now be apparent: We must use NDA techniques that rely on only *one* interaction in the sample, per particle (photon or neutron). Then, all the information remains known except that one interaction between the particle and the sample. In this way, we can determine what that interaction was and thereby determine the composition of the sample.
The only two techniques that meet this criterion are the NRTA and NRF techniques, which transmit neutrons and photons, respectively, through the sample. The reduction in the beam due to interaction in the sample represents the results of only one interaction per particle. The energy, time, and location of the particles in the beam all remain known. In this way, the type and location (in the area, not volume, of the sample) of the interactions in the sample can be determined, and thus, the composition of the sample can be deduced.
Hereafter we describe roughly about two possible NM accountancy measurement techniques: NRD and NRF using intense LCS γ-rays (mono-energetic γ-rays).

## 3.1 NRD (Neutron Resonance Densitometry)

In NRD [1][4][5], neutrons from a pulsed neutron source pass through the sample objects (debris) and are counted with a neutron detector. The resultant transmitted neutron spectrum brings a lot of information about the nuclear material isotopes in the objects. Figure 6 shows the total cross sections of uranium and plutonium in the energy range of 0-30 eV. The grey line in Figure 6 is an example of a transmission spectrum from a NM sample. Such spectra give us precise information about the quantities and ratios of the NM isotopes in the debris. This technique could be applied to objects in thin, slab-type containers (~1 cm thickness or less, for example) or to small samples. Another TOF technique concerned with NDA is NRCA. Gamma-rays emitted during neutron-capture reactions in a sample are measured. From the time-dependent gamma-ray energy spectra, information about isotopes with large neutron-absorption cross sections is obtained.

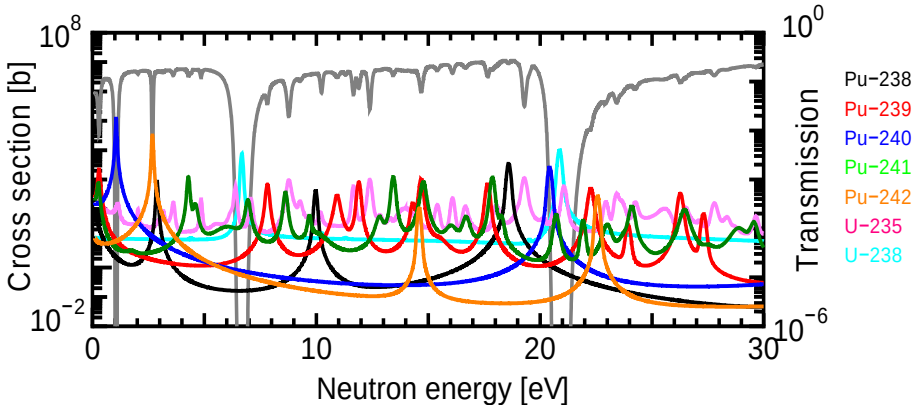


Figure 6 Energy dependences of total cross section of uranium and plutonium (left scale). The gray plot is the neutron transmission of a fuel sample.

We here propose “Neutron Resonance Densitometry (NRD),” which is a combined technique of NRTA and NRCA, for characterization of particle-like debris. The conceptual view of the practical NRD system is given in Figure 7. We have already applied the NRTA and NRCA methods in other studies, such as studies of nuclear data, using the pulsed neutron sources at J-PARC in JAEA and at the KURRI linac of Kyoto University. JAEA and JRC/IRMM have started collaboration studies on Neutron Resonance Densitometry since May 2012. GELLINA of JRC/IRMM has a large scale TOF experiment facility and has been used for NRTA and NRCA. The first stage of the collaboration is on R&D of the NRD method, where the various origins of systematic uncertainties will be studied. A prototype NRD system will be also developed at the JAEA site using a D-T neutron source.

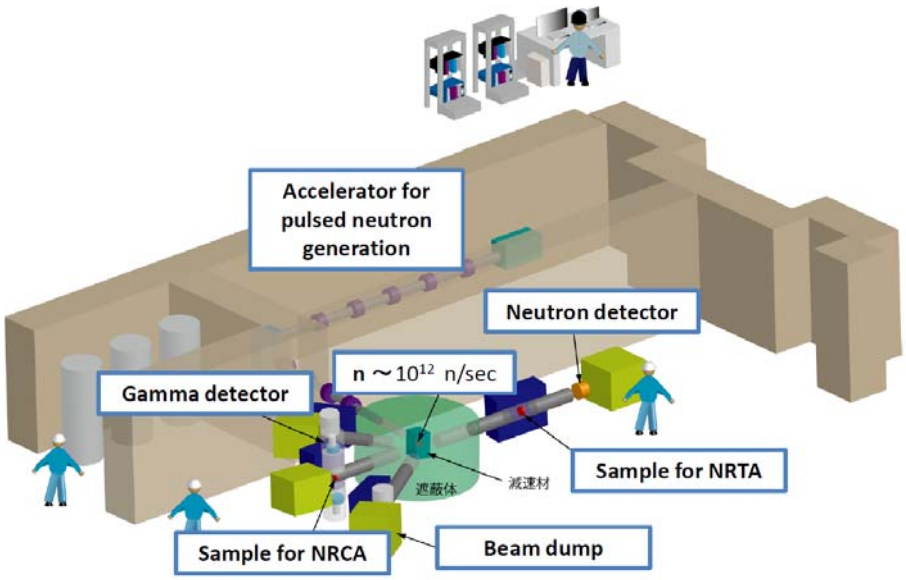


Figure 7 A Schematic View of the NRD System

## 3.2 NRF using intense LCS γ-rays (mono-energetic γ-rays)

An active photon (gamma-ray) technique that uses intense, high energy (2 - 5 MeV), mono- energetic gamma-rays that are tuned to the nuclear resonance states of specified NM isotopes could be an promising technique for the measurement of NM in the debris. (See Figure 8.)

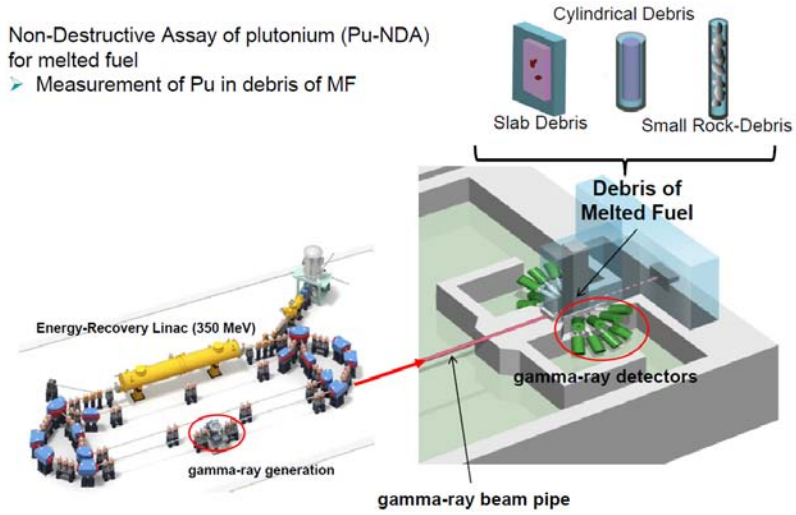


Figure 8 A Schematic View of the NRF NDA System using intense LCS γ-rays (Mono-energetic γ-rays) with Scattered Gamma-rays Counting Method

We JAEA propose two basic methods for this technique [2][6][7][8]. One is measuring the NRF scattered gamma-rays (the NRF scattered gamma-rays counting method), and the other is measuring the transmitted gamma-rays (the transmitted gamma-rays counting method or witness foil (plate) method).

## 3.2 (1) NRF Scattered Gamma-rays Counting Method

The scattering of the interrogating mono-energetic gamma-rays (which are tuned to the excited energy states of a specific isotope of NM, such as $^{239}$Pu) by the process of nuclear resonance fluorescence brings us information about the quantity of the specific isotope in the MF debris. By counting the

scattered gamma-rays that have the same energy as the interrogation gamma-rays, we can quantify the amount of the specific isotope (such as $^{239}$Pu). Figure 9 shows how to make a measurement of a specific isotope with this method.

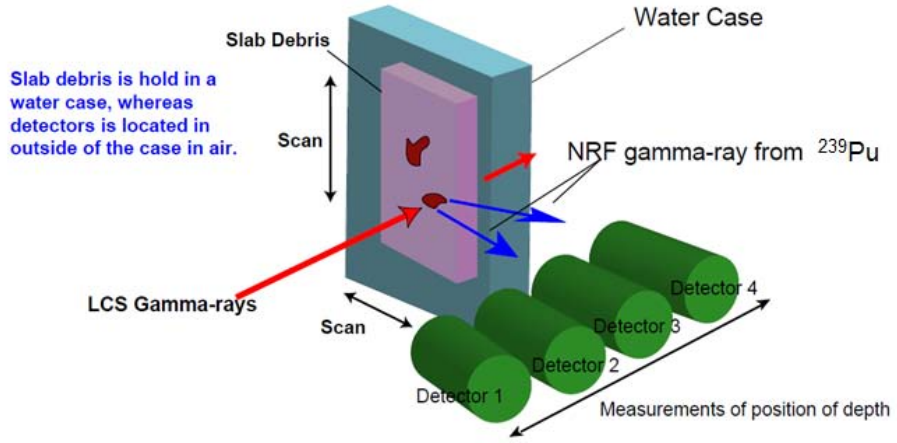


Figure 9 NRF Scattered Gamma-rays Counting Method

## 3.2 (2) Self-interrogation (or Witness Plate) Method

Measurement of the $^{239}$Pu (for an example) in the melted fuel is possible by using the γ-ray self-interrogation (or witness plate) method [6]. It consists of measuring the nuclear resonance absorption (NRA) of photons in the beam. The method works because when a beam of photons traverses the melted fuel, the resonances in $^{239}$Pu deplete the beam at only the specific energies that are characteristic of $^{239}$Pu. The NRA signature can be detected by measuring the decrease in the γ-ray signal from a separate target (witness plate) of $^{239}$Pu placed after the melted fuel debris. The decrease in signal intensity is proportional to the mass of $^{239}$Pu in the melted fuel. (See Figure 10.)
The γ-ray self-interrogation method is insensitive to the composition and geometry of the melted fuel fragment. As γ-rays interact with the melted fuel, they are removed from the beam path and decrease in energy. The γ-ray beam emerging from the fuel has been linearly attenuated by the fuel. The attenuation from atomic scattering (arising from composition and geometry) is corrected by measuring the flux before and after the beam intersects the melted fuel. This correction allows the $^{239}$Pu to be accurately assayed for any composition or geometry. The radioactive nature of the fuel is also not problematic as the measurement is carried out by observing scattered radiation on the witness plate, not by directly monitoring scattering on the melted fuel. Secondary scattering of the radiation from the melted fuel on the witness plate is not an issue, as the beam intensity is much higher than the radiation intensity.

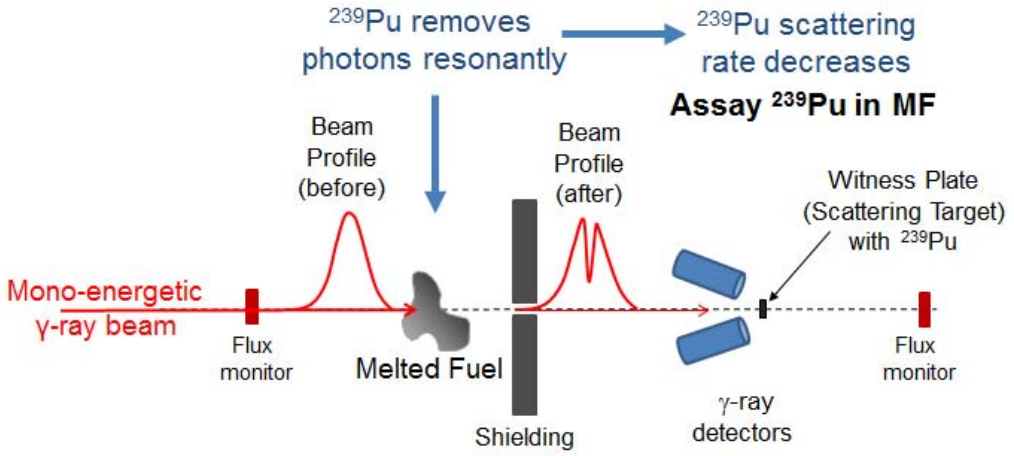


Figure 10 Self-interrogation Method (Witness Plate Method)

Measuring γ-rays with a little bit complicated detector arrays greatly increases sensitivity (1% uncertainty in $^{239}$Pu in about 8 minutes).

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# The 'Room within a Room' Concept for Monitored Warhead Dismantlement

**Jennifer Tanner, Jake Benz,** Pacific Northwest National Laboratory, USA
**Helen White, Sarah McOmish, Keir Allen,** Atomic Weapons Establishment, UK
**Keith Tolk,** Milagro Consulting LLC, USA
**George Weeks,** Savannah River National Laboratory, USA



***Abstract:***

*When considering a hypothetical future monitoring activity to verify nuclear warhead dismantlement, a fundamental chain of custody challenge is to ensure that all the special nuclear material that enters the process is accounted for at the end of the process, without being able to do a true mass balance. It is anticipated that the monitoring party will not be allowed, due to national security concerns, to witness the dismantlement processes.*

*To create a robust chain of custody regime it is necessary to control all the boundaries within the processing area. To simplify the monitoring of these boundaries, one solution is to create a temporary room or volume which encompasses the entire processing area. The walls, ceiling and floor can be controlled without viewing any potentially sensitive dismantling processes. This allows for the detection of unauthorized diversion of the nuclear material.*

*This paper will introduce the 'room within a room' concept, which was created to support a chain of custody based technology trial. Potential candidate technologies will be discussed, as well as lessons learned and potential areas for future development.*

**Keywords:** Nuclear Weapon, Dismantlement, Chain of Custody, Containment, Surveillance

## 1. Introduction

Over recent years, increased attention and emphasis has been placed on investigating technologies that may be used to provide chain of custody of nuclear weapons as part of a dismantlement monitoring regime. Chain of custody, in this context, is an integrated series of procedures and technologies designed to provide:

- Access control over Treaty Accountable Items (TAIs).
- Control over monitoring equipment and data.
- Confidence that no attempt has been made to tamper with, spoof or divert TAIs, monitoring equipment or data.
- The link between the various technical measures required to establish the identity and authenticity of the TAI.

The inclusion of a dismantlement phase within a disarmament process poses a challenge for a chain of custody regime. Host obligations to protect sensitive and proliferative information could lead to the negotiation of compromises with regards to the 'ideal' chain of custody initially envisioned by the monitoring party [1,2,3] (here we consider options where nuclear and non-nuclear weapon states might be members of the inspecting party). These obligations, and possible resulting constraints, must be addressed throughout the dismantlement process. However, the monitoring team must have confidence at the end of the process that dismantlement has occurred, and no diversion has taken place. The differing perspectives of the parties involved influence and drive the design of the final negotiated chain of custody strategy and selected technologies [1,2,3]. The challenge is to create a robust strategy which meets the requirements of both parties.

When considering a dismantlement phase, the monitoring party will be looking to:

- Ensure that all the TAIs and material that entered the dismantlement phase are accounted for at the end of the process.
- Maintain high confidence in any monitoring measurements that might be made during the dismantlement process.
- Maintain the integrity of the monitoring equipment and any data that might be gathered from the chain of custody regime.

The host party will be looking to provide the monitoring party with sufficient access to implement a regime, but will also be required to protect nationally sensitive and proliferative information. This leads to a number of potential constraints and practical considerations that would need to be accounted for within any implemented regime.

- The Host will have an obligation to protect proliferative information relating to the TAIs. Therefore, it is unlikely that the monitoring party would be given full access to the TAI, but is more likely to be presented with a series of containerised objects. The monitoring party is also unlikely to be given access to watch the dismantlement take place.
- As the dismantlement progresses the TAI may change configuration, probably with different container designs deployed in support of each new configuration.
- The Host has a responsibility to maintain the physical security surrounding assets, staff and operations. Note that building a new facility, solely dedicated for the purpose of allowing monitored dismantlement, would be very costly and would not necessarily solve many of the problems posed. Therefore, dismantlement operations might take place in existing facilities which have multiple missions, some of which are highly sensitive. The Host will be looking to protect sensitive information relating to the physical security measures (which could include facility designs) and operations outside the remit of the inspection.
- The verification regime must comply with facility safety regulations. This includes the safety of the monitoring team and Host facility staff. The deployed regime must not restrict the facility staff from safely undertaking all the operations necessary to complete the dismantlement process.

PNNL and AWE are engaged in a technical collaboration investigating chain of custody technologies for use within a verified nuclear weapons dismantlement process. In order to provoke discussion and highlight future areas of research, the team was challenged to design a chain of custody regime which would account for some of the concepts discussed above. The core team was joined by technology developers from additional US laboratories; options were discussed and a proposed solution was trialed. This paper presents the outputs from that technology trial (with particular emphasis on the ‘room-within-a-room’ concept), discusses lessons learned and areas for future development.

## 2. The Trial Scenario

To provide background for the technical trial, a hypothetical scenario was discussed which included a facility, a dismantlement process and a number of restrictions. The technology developers were given the challenge of designing a chain of custody that would account for the TAI through such a dismantlement process, with the given restrictions in place, and to ensure that control was maintained over monitoring equipment and data.

Figure 1 shows a schematic of a hypothetical dismantlement process which has a number of stages [1]; for the purposes of this trial it was assumed that this process took a number of days. The trial focused on a discussion of the chain of custody regime which might be deployed to confirm that this process had taken place. Note that confirming whether the TAI was actually a nuclear warhead was not addressed as part of this trial..

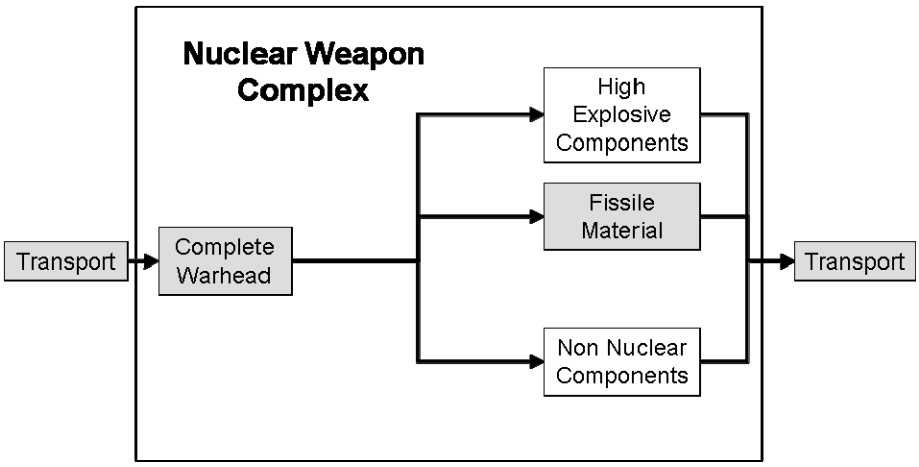


**Figure 1:** Hypothetical dismantlement process [1].

## 2.1. The Facility

The trial was based on a hypothetical bi-lateral, reciprocal, agreement between two fictitious Nuclear Weapon States (NWS) involving the dismantlement of an agreed number of warheads. The notional dismantlement process (Figure 1) included a single stage dismantlement phase, where the TAI would be disassembled into fissile material and explosive elements. A representative facility was used to support the technology trial, which comprised of a corridor, a store area and a simulated dismantlement process area (Figure 2). It was assumed that this facility would sit within a larger complex, housing additional operations outside the remit of the inspection process (Figure 2). Staff and monitoring equipment would enter the dismantlement facility via a change barrier area; the TAI would be transported to the dismantlement facility from a Receipt and Dispatch area. The monitoring party would have access to a work area, outside the high security zone, to review monitoring data.

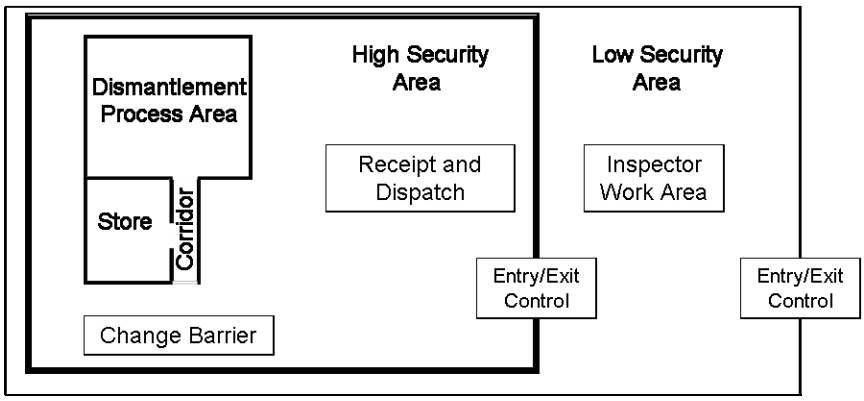


**Figure 2:** Schematic - the representative dismantlement facility within a wider fictitious complex.

## 2.2. The Process

The containerised TAI would be transported to the store at which point a chain of custody trial would begin. The TAI would remain in the store overnight, and then be moved to the dismantlement process area where it would be, notionally, disassembled into fissile and explosives components. For the purposes of this trial it was assumed that this process would take place over several days and would span a weekend. Dismantlement operations would take place during office hours with no work being undertaken overnight or during the weekend. On completion of the dismantlement operations, the containerised components would be moved back to the store area and held overnight, at this point the chain of custody trial would end. Note that the storage periods before and after dismantlement could provide an opportunity to deploy negotiated measurements to establish the identity and authenticity of the TAI [4]; a discussion of such measurements is outside the scope of this paper.

### 2.3. Restrictions

As part of the overarching verification regime, the monitoring party required chain of custody to be maintained for the duration of the process described in section 2.2.. In order to challenge the technology developers, the trial included certain constraints and practical issues, which were imposed by a fictional 'host party'. These constraints were put in place to investigate the impact on a chain of custody strategy.

- The monitoring party was not permitted to remain in the process area while dismantlement was taking place or take images of the dismantlement operations.
- The facility staff required tools to perform the dismantlement operation, and the configuration of these tools could be considered to be sensitive. Therefore, the tools were placed within toolboxes which could not be opened for viewing or sealed by the monitoring party.
- The Host was not able to discuss or reveal aspects of any physical security associated with the facility or transport phases. This means, for example, that only basic schematics of the facility would be available and no measurements of the facility infrastructure would be permitted (other than visible images of non sensitive features). This restriction also extended to the design of the containers which were also be considered a part of the physical security system.
- The monitoring party did not have access to areas deemed to be outside the remit of the monitoring process. The deployed chain of custody equipment could not collect data related to operations being undertaken in other parts of the complex (Figure 2).
- Monitoring equipment would be operated by the Host under Monitor supervision.
- Monitoring equipment would be jointly designed, but built and supplied by the Host.
- The facility staff would require access to power outlets and other services.
- Host fire and safety regulations would need to be adhered to.
- No sensitive information could be released from this monitoring regime. Any data gathered by the chain of custody equipment would need to be reviewed by the Host before it could be released to the monitoring party.

## 3. Developing the Chain of Custody Strategy

Having discussed the background scenario, the technology team discussed and developed a chain of custody strategy to underpin the technology trial.

There are two basic approaches when considering the design of a chain of custody regime: boundary control and 'one-to-one' templating. Boundary control provides evidence of any unauthorised attempt to gain access to sensitive items/areas contained within a designated perimeter. Tamper Indicating Enclosures (TIEs) are a manifestation of a 'controlled boundary' strategy and have an advantage over traditional seals (tamper indicating devices) in that boundary protection is an inherent part of a TIE design [5,6]. A well designed boundary control system integrates basic materials, tamper indicating components and uniquely identifying components (i.e. anti-counterfeiting) into an optimised design.

An alternative approach to creating a controlled boundary is to measure a 'signature' (or 'one-to-one' template) that relates either to the container, the object inside the container or the container/object combination. The signature is then used to maintain chain of custody by either uniquely identifying the item, providing evidence of tampering with the container, or by providing evidence of tampering with the contents of the container [7].

A robust chain of custody consists of a multi-layered, integrated, combination of boundary control technologies and templating concepts. A combination of multiple layers and differing technologies is important in order to ensure (a) there is no single point of failure for the system, (b) there is always redundancy in the system should a component or layer fail and (c) there is more than one source of evidence should an inconsistency require investigation.

In order to complete the chain of custody regime, four further underpinning areas were considered:

1. Threat and vulnerability analysis.
2. Authentication and certification tools, techniques and procedures to ensure that the deployed technologies are trusted by both parties and can be inspected effectively.

3. Data management techniques, procedures and strategies.
4. Technologies and procedures that can be used to guard against ‘spoofing’ or diversion activities. This includes the assessment of:
    a. A facility for background measurements or diversion routes.
    b. Empty containers or containers that are being checked for release from the monitoring regime.
    c. Shrouded items including tooling boxes.
    d. The entry and egress of personnel, containers and equipment.

In the case of the hypothetical dismantlement scenario introduced in section 2, templating could not be employed, as the TAI and container configuration may change as it moves through the dismantlement process. However, boundary control and accountancy methods could be considered. In this approach, the facility would be inspected for potential diversion routes and swept with radiation monitoring equipment to ensure that no sources were present that might be used to ‘spoof’ the regime. Boundary control measures would then be put in place to maintain a monitoring party controlled ‘sterile’ area in which the dismantlement activities could take place. The TAI would be moved in; the monitoring party would then leave the facility so that Host dismantlement operations could take place. Boundary control would be maintained during this phase to ensure that no fissile material could be diverted from the process. At the end of the dismantlement process, all the items would be accounted for, sealed, and removed from the facility. The facility would then be re-inspected to ensure that nothing remained. In theory, everything that entered the area would be accounted for and chain of custody maintained. However, the restrictions imposed by the Host create a challenge to the robustness of this approach. The monitoring party would not have sufficient information about the facility to identify all possible diversion pathways; radiation sweeping would only partially address this. A lack of access to shrouded tooling boxes and empty containers also provides potential diversion routes. This means that even if everything that entered the facility was removed, the monitoring party could not guarantee, with 100% confidence, that all the fissile material resulting from the dismantlement of TAIs was sealed in the officially designated containers.

In order to address the facility inspection issues, the team proposed a ‘room-within-a-room’ strategy. The monitoring boundaries would be simplified by creating a temporary room or volume encompassing the entire process area. The walls, ceiling and floor could then be controlled without viewing any potentially sensitive dismantlement processes or sensitive facility security infrastructure. This would allow for the detection of unauthorized diversion of the nuclear material, or introduction of spoofing materials into the facility. Note that this solution would not account for the potential use of shrouded items or empty containers as a diversion route without additional technical measures.

## 4. Creating a ‘Room-Within-A-Room’

The phrase ‘room-within-a-room’ does suggest the construction of a room sized TIE with walls, flooring and a ceiling. Although this is one approach, the actual realisation of this strategy could take many forms; it is perhaps more appropriate to look at this concept from the viewpoint of boundary control. The version of this concept deployed during this technology trial is summarised in Figure 3 and below.

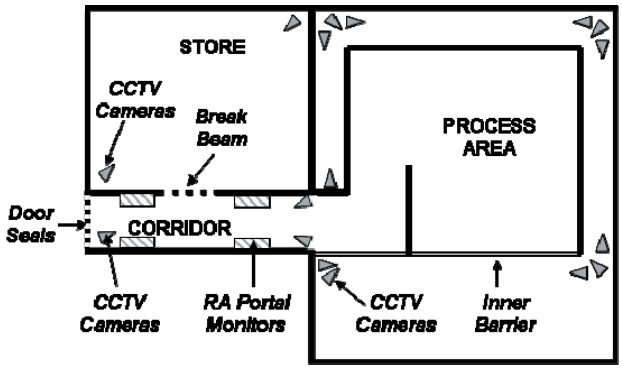


**Figure 3:** The version of the ‘room-with-a-room’ concept deployed for this technology trial.

### 4.1. The Dismantlement Process Area

For this trial, the monitoring party could not be present during the dismantlement operations; neither could any images be taken of the dismantlement operations. This meant that an inner process area needed to be created which would shield sensitive operations from observers, and also provide boundary control, particularly at times when the Monitors would not be present. The 'room-within-a-room' created for this area is summarised in Table 1.

| **Boundary Element** | **Technology** | **Comments** |
|---|---|---|
| Walls | Inner boundary 'walls' created by a combination of temporary barriers, CCTV cameras, image change detection techniques and uniquely identified adhesive seals. | - Barriers to be erected to define the operational area and preclude access to areas that were not required.<br>- A gap to be created between the barriers and the facility wall. Cameras to be positioned to monitor the gap; this would shield the operations from direct view but would capture any attempt to gain unauthorised access to the walls of the facility.<br>- Image change detection techniques to be deployed where the temporary barriers met the floor.<br><br>- Barrier panels to be secured to each other using uniquely identified adhesive seals. |
| Ceiling | CCTV Cameras | - In order to comply with facility fire regulations, the barriers could not go all the way up to the facility ceiling. To accommodate dismantlement operations the barriers could not continue across to form a temporary ceiling.<br>- Cameras to be positioned so that any unauthorised attempt to access the facility ceiling would be captured.<br>- Cameras to be positioned at ceiling height with a restricted field of view which precluded the imaging of any dismantlement operations or physical security features. |
| Floor | Image Change Detection | - Existing facility floor to be inspected for indications of tampering.<br>- Before and after images of significant features recorded for analysis via change detection techniques. |
| Entry/Egress | CCTV cameras, radiation portal monitors, passive sealing system for the door | - Entry and egress to the facility was via the corridor.<br>- The TAI, tooling and equipment to be moved into the facility under Monitor supervision. Once the dismantlement had begun, and the monitoring party was absent, the facility staff would not be allowed to remove these items from the designated process area.<br>- CCTV cameras to be positioned to encompass the key areas of the corridor whilst precluding sensitive physical security features from view.<br>- Portal monitors would measure a gross count for gamma and neutron and would be positioned to capture the movement of sources in and out of the process area.<br>- Any sealing system applied to the door would need to comply with fire regulations, so that the door could be easily opened in an emergency.<br>- Door seals to consist of a passive break bar system (secured with uniquely identified loop seals) and uniquely identified adhesive seals. |

**Table 1:** Room-within-a-room strategy proposed for the dismantlement process area.

### 4.2. The Store

Within this hypothetical scenario, imaging of shrouded TAI containers was permitted in the store area. The chain of custody for the store consisted of a break beam system and CCTV cameras. The break beam system on the door would need to comply with fire regulations so that the door could easily be opened in an emergency; this system would not prevent access, but would provide the monitoring party with a record of an entry event taking place. The CCTV cameras were positioned to cover wherever the containers were to be placed. Again, security features associated with the storage area were to be precluded from images by careful consideration of the camera field of view.

### 4.3. Supporting Hand Held Radiation Sweeping Activities

Gamma and neutron dose meters were deployed to confirm the background radiation values for the facility, and to verify the absence of unauthorised radiation sources either within the facility, associated with shrouded items or inside declared “empty” containers. These ‘sweeping’ activities would be undertaken before the TAI was brought into the facility and after containers associated with the dismantled TAI had been removed from the facility/process area post dismantlement.

### 4.4. Chain of Custody Associated with Monitoring Equipment and Data

Chain of custody over the monitoring equipment would be maintained via a combination of TIEs, seals, and careful positioning of the CCTV cameras. Each CCTV camera was placed into a passive TIE that had been specifically designed for this purpose [6]. All the other supporting equipment, such as digital cameras associated with the change detection activities, were secured in a passive optical fibre based TIE [5,6] when not in use. Similarly, all data recording devices and portal monitors were placed in bespoke designed TIEs. Each camera was positioned so that it was in the direct field of view of at least one other camera. All the equipment to be left within the facility (including all the cabling, data recording devices and TIEs) was positioned in the field of view of at least one CCTV camera.

## 5. Lessons Learned

For this technology trial, the above chain of custody regime was tested in a representative facility for a notional dismantlement process.

### 5.1. The Dismantlement Process Area

The room-within-a-room strategy successfully maintained chain of custody during the dismantlement process whilst effectively shrouding sensitive operations from monitoring party observation. It was agreed that the overall strategy was sound, although the technology used to support it during this trial required further work. There was a heavy reliance on the CCTV system which had the dual function of boundary control and a layer of defence against tampering of monitoring equipment. It was noted that, the loss of the CCTV system would only be partially compensated for by the deployment of TIEs. It was recognized that additional layers of boundary defence associated with the barriers, ceiling and floor were required. Large scale boundary tamper indicating concepts remain of interest from this perspective. The infrastructure required to run the CCTV system for an extended period of time was not exercised during this trial but was discussed as an area that would need to be addressed.

The floor within the process area was a particular area of concern. This was the only surface of the room not covered by the CCTV camera system. There was discussion of putting down a temporary floor which could be inspected. This would have to:

- Meet facility safety regulations.
- Be strong enough to accommodate the weight of the equipment and staff.
- Be robust enough to be able to distinguish between damage from legitimate activity and from tampering.

Although several temporary flooring concepts were discussed (e.g. fibre optics, guided wave, CCTV under-floor imaging) none could be developed in time to support this trial.

Image change detection was used in several ways during the trial:

- Photographs were taken of the unique identifying features associated with the seals. These were either ink runs or simple particle based concepts.
- Wide area reference photographs were taken of the barriers.
- The floor was inspected for significant features and potential areas of tamper; reference photographs were taken for later comparison.

Change detection associated with high resolution images of defined areas, as exemplified by the unique identifiers, worked well. Although the unique identifiers deployed during this trial were very simple particulate tag concepts, a comparison of reference and inspection images clearly identified any areas of damage. It was noted that a secondary layer of defence would be desirable which could be used as additional evidence underpinning any inquiries into how that damage was produced. Taking wide area images of flooring or barriers did not provide the resolution required to undertake this level of analysis; it was too time consuming to take all the high resolution images necessary to cover the area in sufficient detail. Inspecting the floor for significant features was a subjective approach that did not allow for the before and after images necessary to deploy this concept robustly. Future work in this area would be to investigate imaging change detection techniques for large areas.

### 5.2. The Corridor

The design of the door seal was an example of how safety regulations can be a key factor in the specification of a given technology; in this case, the break bar concept deployed was designed to account for fire regulations. The fielded concept provided a sealing solution that could be used when the facility was vacated overnight and over the weekend. This was considered to be a back-up to the main deployment of the CCTV camera system and the portal monitors in the corridor; a passive, unpowered, sealing solution to compliment these active systems. The field of view of the camera system covered the door area and the portal monitors.

There is a human resource issue to be considered when monitoring workers as the recording of images of facility employees may cause some concern. This may have to be accounted for either in the design of the system or in the handling of the recorded data. The use of portal monitors was demonstrated during this trial but research is required into how such a system could be designed so that both parties have confidence in the outputs.

### 5.3. The Store

Along with the CCTV system, a break beam sensor was deployed to cover the door of the store room. Again, this allowed for fast access in the event of an emergency, such as a fire. The team liked the simplicity of this door sealing solution and the ease with which information from the system could be accessed; further work is required to make this a secure, deployable system.

### 5.4. Supporting Hand Held Radiation Sweeping Activities

Hand held dose meters should detect significant radiation hot spots within the facility area. This technique alone cannot account for shielded sources. While the facility boundaries were accounted for by the room-within-a-room concept; the shrouded items and empty containers remained a potential route for diversion of material. Confirming the absence of radiation sources in shrouded items and empty containers remains an area for future development.

### 5.5. Chain of Custody Associated with Monitoring Equipment and Data

The deployment of TIEs decreased the number of seals that were required and, therefore, decreased the overall workload for the monitoring party. This is an active ongoing area of work for this collaboration [5,6].

For this trial, the issue of authentication and certification of equipment was only touched upon. It is important that both parties have confidence that all of the equipment: conforms to agreed specifications, is incapable of divulging sensitive or proliferative information, complies with facility regulations, provides an accurate record of the verification activity and can demonstrate that the ultimate output has not been altered [8,9]. The task of authentication and certification is not always

straightforward, particularly when considering more complex active systems. It was agreed that the development of techniques to support authentication activities is a significant area for future research.

This trial highlighted some of the challenges faced when considering the chain of custody of data within this kind of regime; and this has been highlighted as a future area for research. The volume of monitoring data generated presents a challenge for both the Host and the monitoring party during their respective review processes.

## 6. Future Development

In summary, the trial highlighted the following as potential areas for future development:

- Boundary tamper indicating concepts that have the potential to be scaled to cover room sized areas. These should consider the inspection method which needs to be fast and simple to deploy. Currently, this collaboration is focusing on passive concepts which provide a visual indication of unauthorised penetrations; future work could also include active variants.
- Tamper indicating flooring.
- Fast, easily deployed, imaging change detection techniques capable of covering large areas.
- The use of radiation portals within a nuclear weapon dismantlement scenario.
- Inspection techniques determining the absence of unauthorised radiation sources within shrouded items.
- Improved data management techniques for use within a nuclear weapon dismantlement scenario.
- Improved tamper indicating enclosures for monitoring equipment. This collaboration currently focuses on passive concepts [5,6] but also has a long term interest in active alternatives.
- Authentication and certification of monitoring equipment.

## 7. Conclusion

The team was given the challenge of designing a chain of custody regime for a dismantlement scenario that would account for Host sensitivities with regards to safety, security and non-proliferation, whilst providing confidence to the Monitors that dismantlement had in fact occurred. The regime had to account for all TAIs throughout the process whilst shielding sensitive dismantlement operations from Monitor view. A strategy was devised, called ‘room-within-a-room’, which created a controlled area in which operations could take place whilst still maintaining boundary control. There are many ways to realise such a strategy; one approach was trialed, and discussed here, which focused on the use of temporary barriers, CCTV camera systems, portal monitors and various sealing concepts. This approach successfully maintained chain of custody for the duration of the trial; however, a number of areas for improvement were highlighted.

Overall, this was considered to be one potential chain of custody strategy which might be deployed during a nuclear weapon dismantlement process. Key drivers impacting on the success of the approach are the final negotiated positions balancing Host and Monitor concerns and practical considerations associated with the facility in which it will be deployed. Ultimately, research in chain of custody requires a toolbox of technologies with the flexibility to respond effectively to a number of different potential future treaty scenarios.

## 8. Acknowledgements

The team would like to thank the U.S. National Nuclear Security Administration Office of Nuclear Verification and the UK Ministry of Defence, for their continued support of this technical collaboration. The team would also like to thank the many contributors to the successful trial including the DOE national laboratories and facilities and UK AWE staff.

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# Templating As A Chain of Custody Tool for Arms Control

**Jake Benz, Jennifer Tanner, Leesa Duckworth**

Pacific Northwest National Laboratory
PO Box 999
Richland, WA 99352

***Abstract:***

*Historically, templates have been considered for use as a treaty accountable item (TAI) authentication tool, alongside item attributes. Because of this, the use of templates has fallen by the wayside due to the perceived intrusiveness of and handling/storage of template data; especially when compared to the negotiability of unclassified attribute threshold values. However, as a chain of custody tool, templates potentially have a large and important role to play in maintaining confidence in the authenticity of the treaty accountable items as they progress through an arms control regime. In general terms, templating is the process of creating a unique, measurable, and repeatable signature which is representative of the TAI. At any point in time, the signature can be re-measured or re-inspected to verify the signature has not changed.*

*Chain of custody is the process by which a controlled boundary is established and maintained around a TAI to both deter and detect unauthorized access to the item. Typically, this is accomplished by putting a tamper indicating device (TID) on the item or container. The TID now acts as a surrogate for the item itself, and is continually checked to ensure the unique identifier and tamper indicating mechanisms have not changed since last inspection. This in and of itself is a form of templating. A stronger template is one that utilizes a signature of the combined item and container. There are many potential signatures which may be exploited, including radiation-, electromagnetic-, and acoustic-based signatures.*

*This paper/presentation will explore the technology and mechanisms in which templating can be applied to create a more robust chain of custody over treaty accountable items as part of a future arms control regime.*

**Keywords:** Chain of Custody, Templates, Nuclear Warheads

## 1. Introduction

> *"Chain of custody aims to track the movement of the warhead and its components from the point where authentication is established until final disposal or storage….chain of custody creates a controlled environment for the warhead and its components as they proceed through the different stages... This surrogate environment is porous enough to allow controlled entry and exit of items necessary for dismantlement…"*[1] *(VERTIC 2009)*

In the context of arms control regimes, chain of custody (CoC) is the process which allows a party to maintain confidence in the authenticity and integrity of an item. Chain of custody must be maintained over any item which will be used toward treaty verification. This includes the treaty accountable item

[1] Verification Research, Training and Information Centre (VERTIC), "Disarmament: security context and verification challenges," *Trust & Verify, VERTIC Newsletter, Issue 126, July-Sept 2009.*

(TAI), equipment, tags/seals, and even data. What CoC does not do is establish authenticity; this is accomplished through authentication and initialization or confirmation. Authentication is defined as "the process by which the Monitoring Party gains appropriate confidence that the information reported by a monitoring system accurately reflects the true state of the monitored item"[2], and would be used to establish confidence in equipment or data integrity. Initialization or confirmation, the introduction of the TAI into the treaty regime, would be performed on TAIs utilizing techniques such as non-destructive assay (NDA) or non-destructive evaluation (NDE). While chain of custody must be maintained over all equipment which will be utilized by the monitoring party within a treaty regime, this paper will focus only on the use of templates for chain of custody of the TAI.

In order to be effective in maintaining confidence, CoC must begin when an item is introduced into the treaty regime, and be maintained throughout the item's lifecycle or the lifetime of the treaty. For example, in a warhead monitoring regime, this would require providing CoC of the TAI (warhead) from field deployment through its lifecycle of transportation, storage, maintenance, retirement, dismantlement, and disposition. How the TAI is introduced into the treaty regime, or warhead confirmation, is a key component of chain of custody and can occur either immediately preceding, or any time after, establishing chain of custody. Typically, this is accomplished by putting a tamper indicating device (TID) on the item or container. The TID now acts as a surrogate for the item itself, and is continually checked throughout the warhead lifecycle to ensure the unique identifier and tamper indicating features have not changed since last inspection.

Item initialization or warhead confirmation, verifying the warhead is as is declared, typically includes either attribute and/or template measurements to provide initial confidence that the item declared to be a nuclear weapon, or component of, is as declared. Attributes are physical characteristics of the item that can be measured and quantified. Examples include the age of plutonium, isotopic ratio of Pu-240/Pu-239 less than or equal to 0.1, or a mass of at least 500 grams[3], and represent unclassified threshold values that can be used to claim that a presented item is a weapon, or is of weapons-quality or origin. These examples represent just a few of the potential attributes that could be used. The reason attributes are popular with negotiators is that the results are unclassified and usually verified simply by a red or green light indicating the measured item has passed the given attribute test.

Templates measure the item to the highest degree of fidelity possible for the given equipment. Example templates can be measurements of the radiation spectrum emitted by the item, an image of the item, an electromagnetic signature, or any other signature intrinsic to the item. Any change in configuration, removal or introduction of material (shielding or fissile) should result in a non-match to the original template (golden copy) when the item is re-measured. There have been traditionally two main arguments against the use of templates. The biggest concern regarding templates has been that they require the long-term storage and management of highly classified information. This made the use of templates more difficult to negotiate, however they provided a much higher confidence in the authenticity of the item when compared to the "golden copy". The second downside of the template approach for item initialization was that the monitor only had to be fooled once during the creation of the golden copy reference measurement (the "initialization problem"). The initial reference measurement is the base against which all other measurements are compared. If this measurement could be modified or substituted to the host's advantage, then the confidence of the whole verification regime has been compromised.

In the context of chain of custody, templates may find use in ways which simultaneously increase confidence in the treaty regime, while alleviating many of the traditional concerns with the use of templates for warhead confirmation. The goal of CoC is to maintain the integrity of the declared item from unauthorized substitution or diversion, not to prove authenticity which is the goal of warhead confirmation. In the case of CoC, it matters less what the measured item is, as long as the template remains unchanged between measurements. Additionally, in a CoC application, the template is only associated with an individual item as it traverses its lifecycle and the treaty regime. Once the item leaves the regime, or changes configuration, the template is no longer valid and can be destroyed. This eliminates the "golden copy" issue since the template will not be used as the basis for which an entire class of weapons will be compared. And it also alleviates the need for long term storage of classified information, over the lifetime of the treaty, since the monitors only require the template while

[2] Richard T. Kouzes, "A Dictionary for Transparency," Pacific Northwest National Laboratory, PNNL-13723, 2001.
[3] Richard T. Kouzes and Bruce D. Geelhood, "Methods for Attribute Measurement and Alternatives to Multiplicity Counting," Pacific Northwest National Laboratory, PNNL-13250, 2000.

the item remains in the same configuration in which the template was generated. An additional benefit of utilizing templates for CoC is the potential ease and rapidity of producing templates, in many cases on the order of a few seconds to a few minutes. Since there is no need to authenticate the item, there is no longer a need to analyze or visualize the measurement results, therefore, it eliminates the requirement to develop complex analysis software. This approach differs from that of attribute measurements which must have sufficient resolution and statistics, as well as robust analysis software, to verify with confidence, that the negotiated thresholds have been met.

It is worth noting, that certain types of templates offer the potential to further alleviate the need to store classified data. There are two avenues through which this can be pursued. The first is through the generation of unclassified templates. Further investigation is required, but certain technologies discussed in the following sections may actually perform measurements which provide unclassified results. The second is through the one-way hashing of classified data. The hashing algorithm will encrypt the data and produce an unclassified digest which can be generated and compared during subsequent measurements[4].

A template measurement utilizes the intrinsic signature from the item itself, but also incorporates signatures from the container and the configuration (material and geometry). This is because the template signature is affected not only by the warhead, but also how the warhead signature interacts with or is changed by the intervening material and container before it is measured. Therefore it becomes an aggregate signature which is stronger than the sum of its parts and can act both as a unique item identifier, as well as a tamper indicating feature. This perspective differs significantly from attribute measurements which consider container effects to be a nuisance and a detriment to a good measurement result.

## 2. Potential Technologies and Signatures

There are many areas where templates are already being used as a chain of custody tool. Tags and seals are routinely verified by checking that the unique identifier is the same as last time, and that the tag/seal has not been tampered with. In effect, the inspector is performing a template comparison by confirming the tag/seal has not changed since last inspection. Static objects, equipment, and even rooms are also verified using templates by performing change detection on photographs taken of the items/rooms and compared to reference inspection photos. Finally, data, including software and even hardware, is commonly authenticated through the use of templates. Hash functions and digital signatures are utilized to protect data from unauthorized redaction or substitution. They are a powerful form of template measurements where one byte difference substantially changes the hash function or digital signature. Authentication is accomplished through the use of public-private key encryption by comparing hash digests.

There are two main CoC technology areas which may find use for templating and accounting of TAIs, nuclear radiation measurement methods and non-nuclear measurement methods. Under radiation-based templates there are two subcategories, gamma and neutron. Gamma and neutron radiation detection is a very mature field, and there are many tools available for template generation. Non-nuclear measurement methods also contain two broad subcategories, electromagnetic and acoustic. Both areas are well-developed and commonly used in other fields, but have only limited success in the area of arms control and chain of custody.

<u>Nuclear Radiation Measurement Techniques</u>
Gamma detectors run the gamut from low (plastic scintillators) to high resolution (high purity germanium), and each have benefits and drawbacks for CoC application. High purity germanium (HPGe) detectors make-up the highest resolution detectors and allow for the creation of a detailed and complex radiation template which would be very difficult to spoof. On the flip side, the level of complexity may make consistent replication of the template signature difficult. Small changes in background or even in the orientation of the item within the container, may lead to a variation in the

[4] Ken Jarman, Sean Robinson, Allen Seifert, Benjamin McDonald, Alex Misner, Tim White, Erin Miller, and William Karl Pitts, "Non-Invertible Transforms for Image-Based Verification," Proceedings of the 52nd Annual INMM Meeting, Desert Springs Palm Desert, CA, PNNL-SA-80555, June 9, 2011.

radiation field which could result in a non-match to the original template. On the low end of the resolution spectrum lay plastic scintillators, which find use in a wide range of applications including homeland security. Normally, these detector are utilized as gross counters, but can be configured to include crude spectroscopy by incorporating 3 - 4 energy windows. While efficient, the lack of resolution makes spoofing significantly more feasible for an adversary. In between these two extremes are a suite of detectors which can be classified as medium resolution, with typical resolutions somewhere near 5% - 7%. Examples include Sodium Iodide (NaI), Cadmium Telluride (CdTe), and Cadmium Zinc Telluride (CdZnTe). These detectors offer the benefit of operating at room temperature and are more efficient that HPGe detectors. NaI has already been developed and tested as a templating system under previous work preparing for START III negotiations. This system is called the Trusted Radiation Identification System (TRIS) and was developed by Sandia National Laboratories.[5]

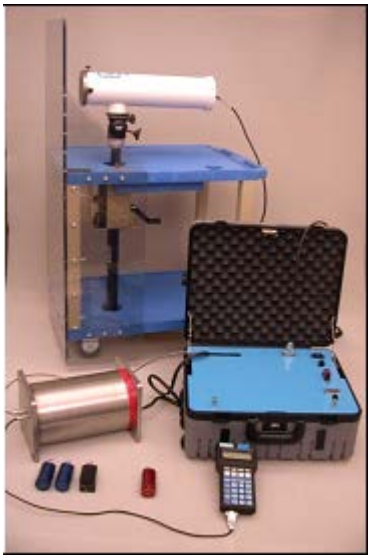


**Figure 1: Trusted Radiation Identification System (TRIS)**

There are also a whole suite of neutron detectors available, the majority use helium-3 which is a very good thermal neutron detection material, but is rare and expensive. Recently there has been a push to explore alternative materials (Kouzes et al. 2010). The benefit to using neutrons is that they are not easily shielded and readily escape the item and container. Additionally, while the fission neutron spectrum would be fairly consistent between weapon types, the quantity of neutrons emitted and the interaction of the neutrons with the intervening material will most likely create a unique signature for different weapon types, or even same weapon types in different configurations. In fact, neutron template measurements have the distinction of being the only negotiated use of radiation measurements utilized in a U.S. treaty. In the Intermediate-range Nuclear Forces (INF) bilateral treaty with Russia, a neutron template measurement was performed and implemented to distinguish between the allowed SS-25 ICBM and the to be eliminated SS-20 ICBM. The template utilized within INF was simply the gross neutron count rate measured across a grid. The variation over the grid determined whether or not 1 or 3 warheads were contained in the nose cone. Gross neutron count rates, or even simple dose measurements would provide a robust template as they are dependent on the quantity and configuration of fissile and other intervening material. In the U.S., the combined gamma-neutron dose rate of a nuclear warhead is unclassified.

Imaging could provide a very powerful template, and can be utilized through both radiation and non-nuclear measurement methods. Passive imaging would generate a template of the intrinsic radiation signature of the special nuclear material (SNM) and its interaction with intervening material. Variations of either will result in a different image upon inspection, and would not match the reference

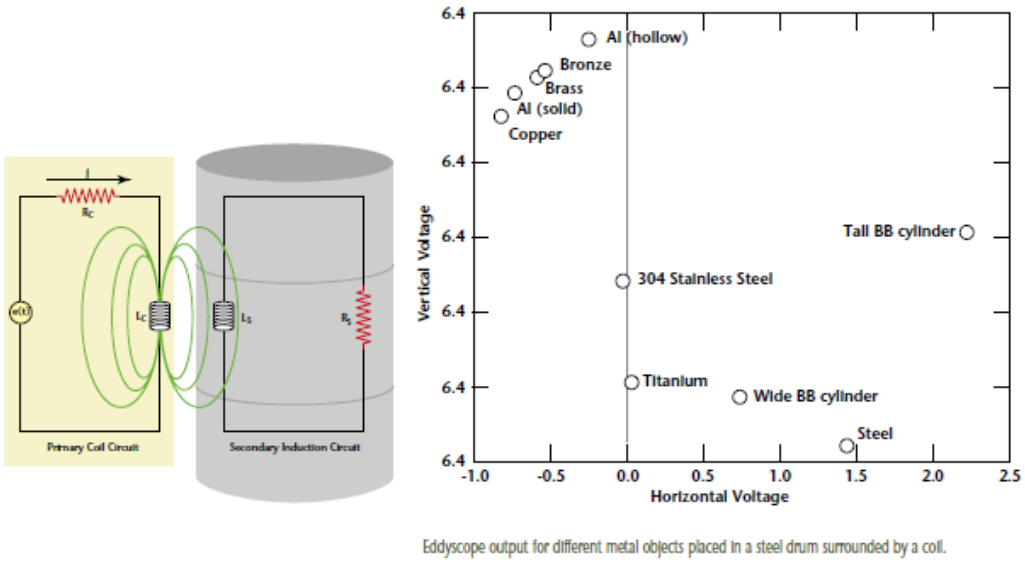
[5] K.D. Seager, D. J. Mitchell, T. W. Laub, K. M. Tolk, R. L. Lucero, and K. W. Insch *Trusted Radiation Identification System*, Proceedings of the 42nd Annual INNM Meeting, Indian Wells, CA, 2001.

template. Example techniques include autoradiography, coded aperture imaging, Compton scatter imaging, gamma-ray imaging, and neutron elastic scatter imaging. In the event the item is a weak neutron or gamma source, or the emitted gamma-rays are low energy and sufficiently shielded, the radiation may be stopped prior to reaching the imager. In these cases, it may be necessary to introduce an active interrogation source. An active interrogation source, gamma or neutron, can induce additional fission events in the item. In the same manner as above, this will provide a template of the source distribution and the configuration of the intervening material. With the introduction of an external source, it becomes possible to perform both transmission and scatter imaging. Examples include gamma, neutron, and x-ray tomography, fission mapping, and neutron backscatter. The technologies interact differently with the variety of materials which may be present in the item. Gammas and x-rays will indicate the presence and extent of high-Z material, while neutrons are better suited for detecting the presence and extent of low-Z material.

Non-nuclear Measurement Techniques

Infrared (IR) imaging is an example of a non-nuclear technique for imaging. IR imaging utilizes the heat signature resulting from the fission process and energy deposition/escape of fission products as they interact with the nearby material. The heat output profile from the container or body caused by this process may potentially have a unique signature which could be used to generate an image template.

Electromagnetic methods provide the potential to perform "whole container" measurements which incorporate both the container and the object inside. The EM Coil is a technology originally developed by Pacific Northwest National Laboratory (PNNL) in the 1990's that has been resurrected again with the renewed interest in arms control. EM Coil operates using a low frequency magnetic field which can penetrate the container and interact with the metallic object inside[6]. For the same container and different metallic objects inside, either size or material, the coil impedance (template measurement) is different. For different types of containers and the same object inside, the coil impedance is different. This would require any spoof to utilize the same container and an object of the same size, shape and material. Another benefit of this method is that measurement time is on the order of seconds, resulting in minimal disruption of host operations.




Eddyscope output for different metal objects placed in a steel drum surrounded by a coil.

**Figure 2: EM Coil Measurement Results**[7]

Templating containers and rooms pose another challenge which can be addressed through the use of non-nuclear technologies. In these cases, it is impossible to use radiation measurement technologies

[6] A. Mark Jones, Kyle Bunch, Pamela Aker, "Simulation and Experimental Validation of Electromagnetic Signatures for Monitoring of Nuclear Material Storage Containers," Journal of Nuclear Materials Management, Winter 2013 Vol. XLI, Num. 2, 4-13.

[7] Arms Control and Nonproliferation Technologies, "Technology R&D for Arms Control," Spring 2001.

as no fissile material is supposed to be present. Eddy Current is also a technique utilizing electrical impedance and can be used on metallic containers consisting of cold worked austenitic stainless steel, or other metallic alloys with magnetic permeability. This technique is also applicable on welded containers. The welds are sufficiently unique to provide a robust template when measured using Eddy Current[8].

Photography is an electromagnetic technique which finds many obvious uses as a chain of custody tool. The method to point out here is how it can be used to template a room or facility. Reference images can be captured of walls, barriers, doors, pipes, etc, and these form the original template to which future inspection images can be compared against using change detection software.

Acoustic methods also have the ability to generate "whole container" templates. Vibrometry, or vibroacoustics, is a series of techniques which produce an acoustic resonance in the container or object of interest. The resonances can be created by mechanical means, e.g. hitting or pinging the container, or through non-contact means such as a laser[9]. The measured resonance spectrum (template) is highly dependent on a number of factors including mass, geometry, material, and orientation. Small changes have the potential to create a large change upon re-inspection. These types of techniques are commonly used for non-destructive evaluation of objects ranging in size from microscopic to airplane bodies, and are well developed.

Ultrasonics and guided waves also have the capability to template containers, walls, and other wide-area barriers. Guided waves are commonly used to verify the integrity of large objects such as long length, hard to access, pipes in nuclear reactor facilities. Ultrasonic waves are introduced into the object being interrogated through a transducer and are guided through the wall of the object or container until it reaches a wall or defect, and a portion of the signal is reflected back to a receiver. The time of flight, presence of defects, length of the interrogated object, and the type of material all play a part in generating the unique acoustic signal (template) measured by the receiver.

The Ultrasonic Intrinsic Tag (UIT) is a technology which began development in 1989 and was envisioned for use under START where it was originally designed to uniquely identify the skirt of SS-20 and MX missiles. Later versions were modified to interrogate metallic components, such as an Air Launched Cruise Missile (ALCM). The UIT interrogates the subsurface microstructure of the material, at a predetermined depth, by scanning over a specific area of the object with an ultrasonic field. Due to the nature of the subsurface microstructure, portions of the ultrasonic waves will be deflected, some back to a receiver, and result in a unique reflectance pattern. Additionally, since the interrogation area is subsurface, it is highly resistant to tamper or substitution. In 2003, PNNL performed a proof-of-concept test to determine whether the UIT could uniquely identify an item in a blind test. Five high fidelity B61 trainers were presented for UIT measurement and reference measurements were taken. One trainer was chosen at random to be returned for a second measurement, and re-interrogated. Based solely on this UIT measurement, a successful identification was achieved when compared to the reference measurements. The figure below shows one such measurement and the output of a UIT measurement[10]. As a template measurement, the UIT utilizes the intrinsic features of the subsurface of any object, item, or container under interrogation. These features are unique (estimated to be one out of a million), not only to the object, but to the specific location where the measurement was performed. A UIT covers an area of approximately 5 mm by 5 mm and must be re-measured at the same location to within 10 percent[11].

[8] Keith Tolk and Gerald Stoker, "Eddy-Current Testing of Welded Stainless Steel Storage Containers to Verify Integrity and Identity," Sandia National Laboratories, SAND99-1829C, 1999.
[9] http://www.polytec.com/us/products/vibration-sensors/. Accessed May 1, 2013.
[10] M.S. Good, B.E. Simpkins, J.R. Skorpik, J. L Kirihara, J.A. Willett, "Ultrasonic Intrinsic Tagging for Nuclear Disarmament: A Proof-of-Concept Test,", Pacific Northwest National Laboratories, PNNL-14462, Oct. 2003.
[11] M.S. Good, J.R. Skorpik, L.J. Kirihara, K.L. Gervais, J.E. Tanner, "Ultrasonic Intrinsic Tagging For Unique Item Identification," Institute of Nuclear Materials Management 49th Annual Meeting, PNNL-SA-61396, 2008.

**Figure 3: Model UIT-5100 applied to an inert B61 surrogate[12]**

The above nuclear radiation measurement and non-nuclear measurement methods provide a general overview, and introduce a few specific examples, of the wide variety of technologies which could be used to produce templates within a chain of custody regime. There are many other potential technologies which also have the ability to generate robust templates. The following sections will highlight some of the benefits, issues and concerns associated with these technologies as they might be used within the context of a warhead monitoring regime.

## 3. Application Within A Chain of Custody Regime

Within any potential warhead monitoring regime, there are two broad categories under which a warhead could be classified within its lifecycle: deployed or non-deployed. Non-deployed status can be broken up into four additional lifecycle stages: storage, transportation, maintenance, and retirement.

Deployed status means that the warhead or bomb is mated to its delivery vehicle and therefore falls under the purview and accounting rules of New START.

Non-deployed nuclear weapons have, to date, not been captured under any treaty and where a robust chain of custody utilizing templates will likely have the greatest impact.

Storage may include warheads in long-term storage, such as war reserves, or staging while awaiting maintenance or redeployment. As the name implies, the items in storage or staging are unchanging, and there should be minimal fluctuation in inventory or arrangement. Additionally, the items of interest for chain of custody will most likely all contain fissile material. Therefore, any of the technologies described above could be utilized to generate template measurements. Re-inspection frequency can be low since the items and inventory should be fairly constant.

Transportation will be a short-term status as the items travel from one location, and perhaps one status, to another. Examples include movement of items from a deployment site to a different site for maintenance or other life extension projects. The time, duration, and travel path of these item movements are classified, which makes chain of custody very difficult. In this case, the only potential

[12] M.S. Good, B.E. Simpkins, J.R. Skorpik, J. L Kirihara, J.A. Willett, "Ultrasonic Intrinsic Tagging for Nuclear Disarmament: A Proof-of-Concept Test,", Pacific Northwest National Laboratories, PNNL-14462, Oct. 2003.

option may be to verify the integrity of the items prior to, and then post, transport. Templates are ideally suited to this situation, and can provide much higher confidence that an item has not been tampered with or substituted during shipment. Again, the items of interest will most likely all contain fissile material, so either radiation-based or non-nuclear template measurements could be used.

Maintenance may represent the most difficult stage for any chain of custody regime. By definition, free access to the item must be allowed and therefore any traditional means of protection, e.g. tags, seals, tamper indicating enclosures, must all be removed. Additionally, due to the nature of maintenance and life extension, what goes in may not be exactly the same as what comes out. This potential change in configuration can pose a problem for many template technologies, and careful thought must be given to where and how a robust template can be generated. One solution might be to identify a critical component which is never removed or modified, and take a UIT template to uniquely associate that component with the item. In many cases it may still be feasible to utilize radiation templates, where replacement with identical components represents the extent of maintenance activities.

Dismantlement has been the recent focus of a variety of DoE sponsored projects and collaborations, and is an areas ripe for future research and development. The unique aspect of dismantlement is that, at this stage, the items will change configuration. During the dismantlement process, one full-up weapon enters, and multiple containerized objects exit. It may not be possible to utilize one template throughout the entire process. Any change in configuration, i.e. dismantlement, will most likely require a new template. An additional level of difficulty is introduced by the fact that the entire process is highly classified, and therefore inspectors will not be allowed to watch or even look inside many of the containers, empty or full. Facility constraints severely restrict what the inspectors are allowed to do, and even what equipment is allowed in. To accommodate all the restrictions, technologies must be simple, robust, safe, and flexible. The black box which encompasses the dismantlement process might also include templates of rooms, walls, or other controlled volumes. Photography and change detection are examples that have been examined through an ongoing collaboration, examining potential dismantlement chain of custody technologies, between PNNL and the UK Atomic Weapons Establishment (AWE)[13]. Many of the items and objects of interest to chain of custody within the dismantlement process may not contain fissile material, and therefore, a combination of radiation and non-nuclear methods will most likely be required to create a robust chain of custody.

## 4. Challenges to Implementation

Previous sections have provided an overview of potential technologies and their application for use within a chain of custody regime employing templates. This section will comment on a few of the issues and challenges surrounding the use of templates.

Radiation-based measurement technologies utilize the radiation signature of the fissile material contained in the item, either intrinsic with passive measurements or induced via active interrogation. The intrinsic radiation signature of an item requires a source of sufficient strength and a deficiency in shielding to ensure the radiation escapes the item and/or container, and this is not always the case. For example, U-235 emits a 186.7 keV low energy gamma ray which is effectively stopped by only a few centimeters of lead or other dense material. Neutrons are less easily shielded, but are not always emitted in sufficient quantity. Uranium is again a good example which emits spontaneous fission neutrons on the order of $10^{-2}$ n/s-g[14]. For gross gamma or neutron counting and spectroscopy, uranium poses a significant measurement challenge.

[13] J. Tanner, J. Benz, H. White, S. McOmish, K. Allen, K. Tolk, G. Weeks, "The 'Room within a Room' Concept for Monitored Warhead Dismantlement," To be presented at the ESARDA Symposium 2013 35th Annual Meeting, Bruges, Belgium. 2013.

[14] N. Ensslin, "The Origin of Neutron Radiation," Passive Nondestructive Assay of Nuclear Materials, U.S. Nuclear Regulatory Commission, NUREG/CR-5550, Chapter 11, Table 11-1, 339, 1991.

Active interrogation introduces health and safety concerns which must be addressed prior to use. Imparting energy into a nuclear weapon is of utmost concern to the host, and the response of the weapon to that energy must be carefully analyzed. From an inspector point of view, active interrogation may be the only viable technique to generate radiation-based templates on items too weak or well shielded to exploit the intrinsic radiation signature.

If the template measurement is dependent on configuration, then consistency between measurements becomes a big concern. Room size, proximity to other objects, background variation, and measurement location on the item all contribute to the generation of a template. If these cannot be duplicated between measurements, there is a high probability the templates may not match, resulting in a false negative. This is of greatest concern to the host who does not want to be falsely accused of cheating on a treaty obligation, and may make them wary of template measurements.

Non-nuclear measurement technologies are dependent on the variety of materials present and the configuration and the orientation of the materials. If the container material is too thick, the electromagnetic fields or acoustic waves may not penetrate and interact with the interior item. This could produce a false positive environment where the template is not capturing the whole item, just the exterior which would introduce a potential diversion pathway. Non-nuclear technologies are also dependent on configuration, measurement location, and consistency between measurements. Variations of an item along any axis, including its orientation inside a container, will interact with electromagnetic fields or acoustic waves to produce a unique and robust template. One concern is whether or not that exact geometry can be, or needs to be, replicated for each successive template comparison measurement. If the item shifts, if it is sitting on the ground vice a pallet, if the location is warmer/colder; these are just a few questions which would need to be answered prior to tdeployment of these technologies within a regime.

Non-nuclear template generation of facilities, rooms, and defined volumes also must be carefully evaluated for template reproducibility. For example, if a flexible barrier was used to cordon off a section of a room and happened to be directly below an air vent, change detection might detect a difference between inspections. This could introduce anomalies which might result in false positive, or negative, declarations to be made about the confidence of the treaty regime.

## 5. Summary

Templates provide a robust mechanism for verifying the integrity of treaty accountable items as part of chain of custody regime, where they can be utilized to increase overall confidence in the verification of a future arms control treaty. Many of the perceived weaknesses of templates for warhead authentication do not apply in the context of chain of custody, including the “golden copy” problem and the analysis and long term storage of classified data. Both radiation-based and non-nuclear measurement technologies have the potential to generate robust templates and can be employed to protect treaty accountable items, containerized objects, monitoring equipment, rooms, and entire facilities. Additionally, these technologies have the potential to complement traditional chain of custody measures, e.g. tags and seals, and maintain a controlled boundary around the items when tags and seals must be removed for maintenance or life extension activities.

Since analysis and visualization of the template measurement is not required, additional care must be taken to assess the applicability and compatibility of the technology with the specific chain of custody application to avoid introducing a false sense of security. All parties must understand the limitations and issues surrounding the technologies and resulting template measurements to prevent anomalous events from undermining the confidence of the entire verification regime.

# Laboratory for Nuclear Safeguards, Security and Forensics

**János Bagi , András Kovács, István Almási, Sándor Szabó**

Centre for Energy Research, Hungarian Academy of Sciences
29-33 Konkoly Thege Miklós street
1121 Budapest, Hungary
E-mail: janos.bagi@energia.mta.hu

***Abstract:***

*The Centre for Energy Research (EK) was established in January 2012 on the basis of two former independent institutions, Institute of Isotopes and the KFKI Atomic Energy Research Institute. The Centre is part of the research network of the Hungarian Academy of Sciences and one of the Technical Support Organizations of the Hungarian Atomic Energy Authority. It ensures a professional background providing expert advice, services as well as routine and R&D activities supporting authority functions for the Hungarian Atomic Energy Authority, the International Atomic Energy Agency and the Euratom Supply Agency.*
*To support Nuclear Safeguards, Security and within Nuclear Forensics related activities of the national authorities and international organizations and the development of equipment used for such purposes the Nuclear Security Laboratory of the Centre for Energy Research (EK) is setting up a test and training facility of controlled conditions. In our facility we are going to provide test conditions fulfilling the requirements of International Standards. It consists of 2 laboratories. Both of them are going to be equipped with thermal and humidity sensors, pressure-gauge, and radiation probes. The conditions will be recorded into a central database available for all partners. The laboratory will provide test bed services such as conveyors for dynamic tests and an irradiator for static measurements. A wide range of nuclear and radioactive materials is going to be available at our institute: calibration sources, medical isotopes, fuel pellets, CBNM uranium standard, depleted uranium, etc. Furthermore, it will be equipped with full range of accessories from screwdriver to pulse generator might be needed during installation and testing.*

**Keywords:** nuclear security, nuclear safeguards, nuclear forensics, testing

## The Centre for Energy Research

The Centre for Energy Research (EK) [1] was established in January 2012 on the basis of two former independent institutions; the Institute of Isotopes and the KFKI Atomic Energy Research Institute. The new centre is part of the research network of the Hungarian Academy of Sciences. Although EK has recently formed, the centre benefits from long experience in nuclear safety, safeguards, security and forensics since the two former institutes specialised in these topics for decades.

EK's Mission is:-

- To perform research and development in the field of nuclear science and technology for facilitating the adoption and the safe use of nuclear technology in Hungary
- To participate in international research efforts aiming at the establishing a new generation of nuclear power plants and closing the fuel cycle
- To study the interaction of radiation with matter (including neutrons, gamma-rays and electrons)
- To carry out isotope and nuclear chemistry, chemical analysis by nuclear methods, radiography, radiation chemistry, radiation protection, radiation dosimetry, nuclear safeguards, security and forensics, surface chemistry and renewable energy research.

## The Nuclear Security Department

The Nuclear Security Department consists of four Sections: Nuclear, Dosimetry, Radioactive Material Registry Sections and ICP-MS Mass Spectrometry Laboratory and supports the authority functions for the Hungarian Atomic Energy Authority, the International Atomic Energy Agency and the Euratom Supply Agency.

The research centre of the Nuclear Security Department deals with the topics of nuclear safeguards, security and forensics. In our laboratories both Destructive Assay (DA) and Non-Destructive Assay (NDA) methods are applied for the detection, characterisation and verification of nuclear materials.

The main activity of the research centre of the Nuclear Security Department is Research and Development (R&D). However, this activity is complimented by the provision testing and training services. The department has extensive professional experience and offers an advice service in addition to carrying out routine and R&D activities.

## The SCINTILLA project

EK is an active participant in the SCINTILLA project [2], a European Framework 7 Security Project. The SCINTILLA project has gathered a consortium of scientific laboratories and private vendors in order to develop and introduce to the market radiation monitors of new technologies. The main objectives are to provide i) usage specific detection solutions in the form of devices targeted at difficult to detect radioactive sources and nuclear materials ii) a toolbox of sensors through the development of new and/or improved technologies for the detection of nuclear materials.

The project focuses on five use cases; cargo container screening, vehicle screening, personnel screening, luggage screening and portable devices for use by police in airports. The research challenge includes the creation of an effective substitute for Helium-3 based neutron detectors for the use cases. The project develops a range of scintillation and semiconductor detectors including improved NaI(Tl) spectrometers, plastic scintillators with pulse shape discrimination for the detection of neutrons and the utilisation of CZT sensors for imaging and deployment in low-cost miniature devices. The program iteratively tests and benchmarks these technologies against requirements that have been developed in co-operation with expert users. The results are then available to the expert user community to assess the capability of the systems developed within the program. The project also aims to build a SCINTILLA Partnership Network that extends the test-bed services to a community of stakeholders.

Within the SCINTILLA program EK offers developmental testing through the provision of static and dynamic laboratory test facilities. These take advantage of the wide range of nuclear and radioactive materials available at the centre.

## Laboratory for Nuclear Safeguards, Security and Forensics

Within the framework of SCINTILLA project, a new test and training facility is under construction at EK in order to support nuclear safeguards and security research. The new facility will carry out R&D and will be open to stakeholders working in the same topics. The facility will provide a professional environment for developers and the proper conditions for trainers and trainees.

The facility consists of two laboratories equipped with a calibrated background radiation monitoring and recording system, air-conditioning, temperature, pressure and humidity sensors. The facility will provide the test conditions to fulfil the requirements of International Standards. The conditions will be continually monitored and the data will be recorded into a central database available to all test partners.

The laboratory will provide test bed services for both dynamic and static tests. The dynamic test facility will use a conveyor that is 11 m long with a carrying capacity of 25 kg. The position and the velocity

(up to 2.5 m/s) of the conveyor can be pre-set and recorded into the central database. The radioactive source can also be positioned vertically from ground level up to 2.2 m height with a linear drive. Since the source can be automatically moved in 2 dimensions the conveyor can be used for both carrying out source-passing-through tests and detection efficiency profile acquisition.
A small size PC controlled “mini irradiator” will be used for static measurements. This is under construction and consists of a combined (lead and polyethylene) shielding wall, a source holder, that can house variable gamma and neutron radiation sources and a mechanism to move the source from the loading position to the irradiation position.

A wide range of nuclear and radioactive materials will be available at the Centre. These include; calibration sources, medical isotopes, fuel pellets, CBNM uranium standard, depleted uranium and Pu samples. Furthermore, the laboratory will be equipped with full range of accessories from screwdrivers for installation to a pulse generator for use during installation and testing.

Since sustainable education is fundamental in nuclear security, as emphasised by both the IAEA and the Euratom Supply Agency, EK will provide and host National and International training courses in Nuclear Safeguards, Security and Forensics. The training will consist of both theoretical lectures and hands-on exercises. These will include the following topics: radiation detection, radiation protection, gamma spectrometry, neutron detection, verification and characterization of nuclear material, nuclear security equipment.

## Summary

The Nuclear Security Department of the Centre for Energy Research (EK) was established in January 2012. In addition to focussing on Research and Development, the Nuclear Security Department will provide complimentary activities such as testing, training and expert advice. The provision of these activities is strongly connected and in the longer term will further strengthen the expertise provided by the department. This is because a deep knowledge of the principles of the equipment is needed to deliver effective user training and continued user interaction improves not only the quality of training but also research outcomes that are tailored for use in daily practice. Within the framework of the SCINTILLA project, a new test and training facility is under construction in the EK that will be open to stakeholders. This facility will further improve the Nuclear Security Department research and training activities. The establishment of this facility is the first step of the rejuvenation of the infrastructure at the Centre. Through the support of the Hungarian Academy of Sciences the NDA laboratories of the Nuclear Security Department are also scheduled to be upgraded and refurnished.

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# Attaining and Maintaining a Continuity of Knowledge to Draw Safeguards Conclusions with Confidence

**Robert S. Bean**
Idaho National Laboratory[1]
PO Box 1625, Idaho Falls, ID 83415
robert.bean@inl.gov

**Dianna S. Blair**
Sandia National Laboratories[2]
PO Box 5800 MS 1371, Albuquerque, NM 87185-1371
dsblair@sandia.gov

**Chris A. Pickett**
Oak Ridge National Laboratory[3]
PO Box 2008 MS6166, Oak Ridge, TN 37831-6166
pickettca@ornl.gov

***Abstract:***

*As the 21st century progresses, new nuclear facilities and the expansion of nuclear activities into new countries will require the International Atomic Energy Agency (IAEA) to place an ever higher reliance on attaining and maintaining a Continuity of Knowledge (CoK) of its safeguards information than is currently practiced. Additionally, a conceptual view of where and how CoK can be applied will need to evolve to support improved efficiency and efficacy of drawing a safeguards conclusion for each member State. The ability to draw a safeguards conclusion for a member State will be predicated on the confidence that CoK has been attained and subsequently maintained with respect to the data and information streams used by the IAEA. This confidence can be described as a function of factors such as elapsed time since the measurement, surveillance of attributes, authentication of information, historic knowledge of potential system failures, and the number and type of data collections. A set of general scenarios are described for determining what is required to attain CoK and whether CoK has been maintained. A high-level analysis of example scenarios is presented to identify failures or gaps that could cause a loss of CoK. Potential areas for technological research and development are discussed for the next generation of CoK tools.*

**Keywords:** Safeguards; Continuity of Knowledge

[1] *The Idaho National Laboratory is operated by the Battelle Energy Alliance under DOE Idaho Operations Office Contract DE-AC07-05ID14517*
[2] *Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.*

## 1. Introduction

Beginning with the entry into force of the “Treaty on the Non-proliferation of Nuclear Weapons” in 1970, signatory Member States agreed to declare to the International Atomic Energy Agency (IAEA) the location, quantity, and use of fissile materials under their control. [1] The IAEA in turn is charged with verifying the declaration with the goal of drawing a safeguards conclusion regarding whether the fissile material in the Member State has remained in peaceful use. [2] The IAEA develops a safeguards approach with each country, typically under the auspices of INFCIRC/153, to gather, both via declaration by the State and measurement and inspection activities by the IAEA, the information needed to draw a safeguards conclusion. [3] Under the Additional Protocol, the IAEA has additional authority to access and inspect all aspects of a State’s nuclear activities in order to verify that there are no undeclared fissile materials or activities related to diversion of nuclear material from peaceful purposes. Therefore, the IAEA is tasked with verifying the *correctness* (i.e., declared materials and activities) and the *completeness* (i.e., lack of undeclared materials and activities) of each Member State’s declarations.

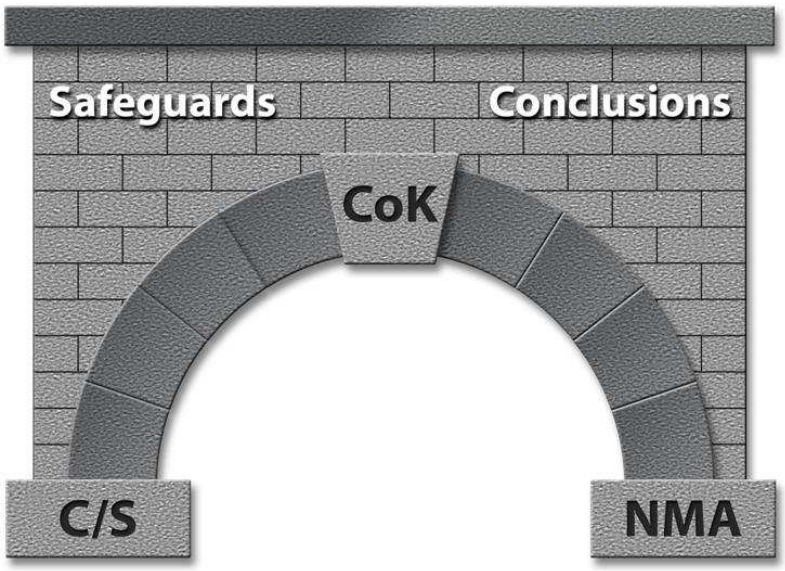


**Figure 1**. Built on a foundation of nuclear material accountancy (NMA) and containment and surveillance measures (C/S), Continuity of Knowledge (CoK) provides the confidence to support a safeguards conclusion. [4]

Historically, nuclear material accountancy (NMA), or rather direct measurement of the nuclear material (via either destructive or nondestructive assay) has formed the basis for safeguards conclusions. For example, a member state’s declarations of material production can be verified by an inspector measuring the product material with IAEA equipment. In addition to NMA, containment and surveillance (C/S), often in the form of container seals and video camera systems, provides a complementary means to verify member state declarations. For example, declarations involving fuel movements can be checked against video footage to determine if the IAEA system saw the same number of fuel elements being loaded, or intact seals can verify that other fuel has not been transferred without being declared. Together, NMA and C/S are used to attain knowledge about the nuclear material activities, and then further used to maintain that knowledge, preferably in a continuous fashion. Continuity of Knowledge for safeguards is the outcome of “a system of data or information regarding an item or activity that is uninterrupted and authentic and provides the IAEA with adequate insight to draw definitive conclusions that nuclear material is not being diverted from peaceful purposes.” [4] In other words, attaining and maintaining Continuity of Knowledge (CoK) is the goal of safeguards measures. If this outcome is achieved, a safeguards conclusion can be supported with confidence.

Note that while the examples of NMA and C/S were used to describe attaining and maintaining CoK, the principle is fully extensible to other types of data that are used to inform the safeguards conclusion. Complimentary access visits under the additional protocol, surveys of literature and publications, and shipping and receiving records, are other examples of means to provide information and therefore contribute to attaining and maintaining CoK. As the IAEA further refines the State Level Approach and moves to safeguard not only newly constructed but new *types* of nuclear facilities (such as long-term geologic repositories), the safeguards approach will require a reliance on information that goes beyond traditional NMA.

The ability of the IAEA to draw a safeguards conclusion about a member state's activities will be predicated on IAEA confidence that the state have been able to attain and subsequently maintain CoK.

## 2. Sampling Theory

Maintaining CoK requires that periodic "checks" be performed to renew confidence that the knowledge is still current and correct. While continuous and/or real-time monitoring are often espoused as the ideal, the reality is that continuous or near-continuous checks generate a tremendous quantity of data that does not necessarily improve the confidence that CoK has been maintained. It is apparent that there must exist some optimum rate that produces the maximum information, or rather confidence that the collected information is correct, without overwhelming both the data collection system and the safeguards inspector who must evaluate the results.

The case of simple production of digital sound files provides an example of sampling theory. [5] A natural signal, from a voice or from a musical instrument is inherently analog. To process these signals in computers, we must convert them to a digital form. While an analog signal is continuous in both time and amplitude, a digital signal is discrete in both time and amplitude. A process called "digitizing," or "sampling" is used to convert a signal from continuous time to discrete time. The value of the original signal is measured, or sampled, at certain intervals in time. The necessary rate of taking samples to ensure appropriate preservation of the original analog information is defined by the Sampling Theorem. The Sampling Theorem simply states that if the original signal contains high frequency components, it must be sampled at a rate higher than the highest frequency component in the original signal to avoid losing information. [6] To preserve the full information of the signal, it is necessary to sample at a rate of twice the maximum frequency of the signal, known as the Nyquist rate. The Nyquist rate gives a reliable benchmark for accurately preserving analog information.

If we sample a signal at a frequency lower than the Nyquist rate, then it will exhibit a phenomenon called aliasing when the signal is converted back into a continuous time signal. Aliasing results from unwanted or

**Long-term fuel storage: A case where CoK will be primary over NMA**

When used nuclear fuel is packaged for long term disposition in a deep geological repository, the nuclear material quantities should be established via NMA. Whether measured by existing techniques, or future advanced techniques [], NMA establishes the inspector's knowledge of the nuclear material quantity. During short term storage and transportation to the geologic repository, additional measures such as C/S maintain CoK such that, if possible, it is not necessary to re-measure the used fuel.

As the geologic repository is filled, it becomes increasingly difficult to perform NMA. Measures such as C/S are used to maintain CoK of the nuclear material. Eventually, the repository will be backfilled, and further NMA will be impossible. Even C/S measures such as container seals will have reduced effectiveness due to the inability to physically verify those that have been buried. Maintaining CoK through measures such as C/S on the repository entrance and seismic monitoring for undeclared digging will become the primary means to maintain Cok and thus support a safeguards conclusion about the nuclear material.

missing frequency components in the reconstructed signal. Examples of aliasing can be seen on automobiles where the wheels of a moving vehicle appear to be turning backwards as the vehicle moves forward. This is due to the fact that your eye is sampling the image at a slower rate (around 10 or 12 times per second) than the vehicle's distinctive rim or hubcap features are moving. The eye, therefore, is losing information because it is sampling too slowly. Conversely, movies appear to display continuous motion because they are filmed at a rate of 24 frames per second—at least twice the rate your eye refreshes—although your eye still will perceive the film of a wheel rotating backward due to the limitations of the eye.

In safeguards, we must ask ourselves, "How much information is needed to maintain CoK?" The sampling rate for checking and re-checking an information stream must be equivalent to or greater than the Nyquist rate for the information signal being sampled. This will be determined by factors such as the type of technology collecting the information, the attractiveness of the nuclear material, the potential diversion or misuse of that material, and the time scale of the diversion or misuse actions. For example, in a situation where low enriched uranium fresh reactor fuel is in storage before being loaded into a reactor, the combination of C/S, possible diversion scenarios (such as diversion to a clandestine enrichment facility and subsequent enrichment to highly enriched uranium), and the safeguards inspection schedule would allow a diversion analysis to be performed on how much time would be required for the operator to divert one significant quantity of U-235 (75 kg of U-235 in the form of enriched uranium enriched to <20%). If we assume that in this case the result of the analysis was that the diversion would take twelve months, then maintaining CoK by "sampling" the facility every six months or less provides assurance that the diversion would be detected without creating an overwhelming demand for either physical inspections or information collection.

Additionally, the concept of nested sampling, familiar to computer programmers, must be considered. For each potential concern, each component will need to be sampled at a minimum rate equal to the Nyquist rate. For example, while it may be necessary for an inspector to visually verify an electronic seal only on a monthly basis, the seal electronics may need to self-test for tamper indications every millisecond and verify communications connections with the data collection system every few seconds. Each part of the system must be sampled, or monitored, such that there is confidence that no information has been lost.

## 3. Maintaining Continuity of Knowledge

Once knowledge has been attained, maintaining CoK is required to have confidence that the knowledge is still correct and, therefore, useable. Continuity of Knowledge must be maintained in such a way that the information is sampled at the appropriate rate to identify issues in a timely fashion. Continuity of Knowledge must be attained and subsequently maintained with sufficient confidence that the knowledge can be used to support a safeguards conclusion. This confidence can be described as a function of factors such as elapsed time since the measurement, surveillance of attributes, authentication of information, historic knowledge of potential system failures, and the number and type of data collections. Additional information, such as indications of attempted tampering, adversary sophistication, and advances in technology will also impact this confidence. This concept is described by the following function:

$$C_{CoK/method} = f(time, sampling\ rate, failure\ rate, adversary\ actions, \dots) \quad \text{(Eq. 1)}$$

where $C_{CoK/method}$ is the confidence that CoK has been maintained by the particular system or method in question.

$$C_{CoK/method} = f(time) \times f(sampling\ rate) \times f(failure\ rate) \times f(adversary\ actions) \times f(\dots) \quad \text{(Eq. 2)}$$

where f() simply represents that $C_{CoK}$ a function of the listed factor. These functions can also be described in terms of confidence that they provide to the overall $C_{CoK}$.

$$C_{CoK/method} = C_{CoK/time} \times C_{CoK/sampling\ rate} \times C_{CoK/failure\ rate} \times \ldots \text{ (Eq. 3)}$$

Mathematically, this function will be multiplicative in nature, because the failure of any component ($C_{CoK/component} = 0$) would drive $C_{CoK/method}$ for the entire method to zero. In practice, the principle of defense in depth leads to the use of multiple, potentially redundant, methods to maintain CoK. In this case the overall confidence that CoK was maintained will be additive as in

$$C_{CoK/overall} = C_{CoK/method1} + C_{CoK/method2} + \cdots + C_{CoK/methodn} \ldots \text{ (Eq. 4).}$$

Note that Eq. 4 is a significant simplification as there may be interdependencies (such as common modes of failure) and no matter how many methods are used total confidence cannot exceed 100%.

In practice, confidence that CoK has been maintained will be highest at the time that a verification action is performed. For an item of nuclear material, this would ideally be quantitative assay. As time passes, confidence that the knowledge in the records correctly reflects the safeguarded item declines. Questions about actions since the last verification - Has the item been moved? Has the seal been broken? Has the unattended monitor failed? Has the operator done something that impacts the item? – can reduce, per Eqs. 2 and 3, the confidence that CoK has been maintained. When the next monitoring action is taken, the confidence is restored to a high level. Failure of a component ($C_{CoK/component} = 0$) will cause a step change to lower confidence that CoK has been maintained. At some reduced level of confidence, it can no longer be sufficiently assured that CoK has been maintained to depend on that knowledge to draw a conclusion. In this case, a full re-verification is necessary to re-establish CoK.

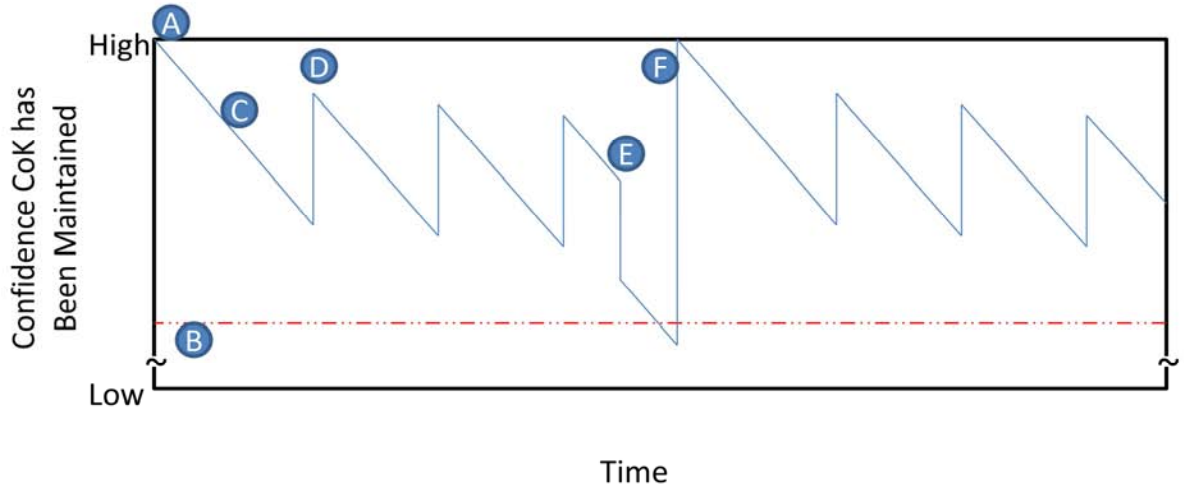


**Figure 2**. Confidence that CoK has been maintained changes over time. A) When knowledge is first attained (through NMA for a safeguarded item), confidence is highest. B) There is some level of confidence below which it can no longer be assured that CoK has been maintained. C) Confidence that CoK has been maintained falls as time passes. D) Sampling the system (such as through a physical inventory, review of unattended monitoring data, etc.) increases the confidence. E) Events such as equipment failure can cause a step change reduction in confidence. F) If the confidence that CoK has been maintained is lost, a re-verification is required.

Previous work has demonstrated this principle, see Figure 3. [7] For both a passive and an active tamper indicating device (TID), the probability of sensing a tamper event was calculated with respect to time. The rising and falling of the probability of detecting a tamper event for the passive TID correspond to

when the device is monitored (i.e., visually inspected) and subsequently left in place. The much smaller fluctuations of the active device are due to the faster sampling rate, allowing the confidence in the TID to be maintained at a high level. In both cases, the probability of sensing a tamper event (in other words, the confidence that the device has maintained CoK of the sealed item) is a function of time, selected technology, and potential adversary actions as described in Eq. 3.

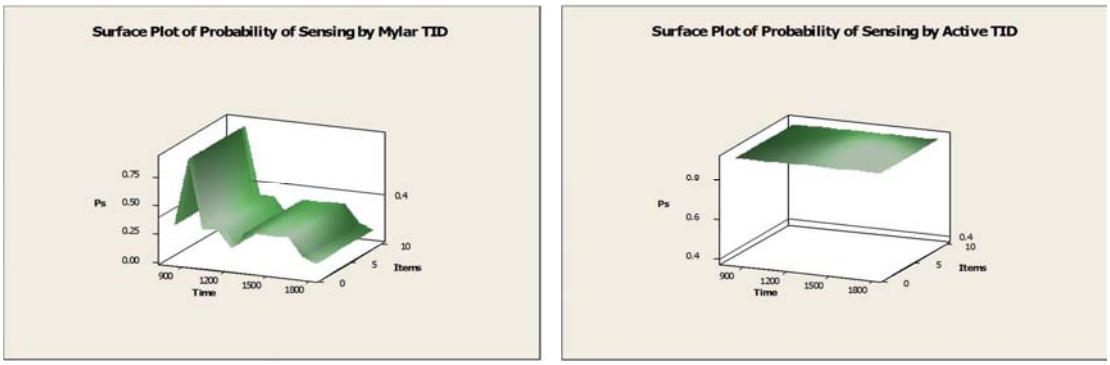


**Figure 3**. Probability of sensing a tamper event for a passive and an active TID. In both cases, the confidence that the TID is working is a function of time and has variations based on rate of sampling the TID status, confidence that it has not been subjected to tampering, etc.

The systems chosen to attain and maintain CoK must, therefore, take into account the principles of

- an appropriate sampling rate for each component, or sub component;
- an appropriate reporting rate for active items (not necessarily the same as the sampling rate);
- the effects over time of confidence in CoK for each component, such as:
  - technology failure rates,
  - effectiveness of technology selections, and
  - adversary capability advances; and
- a systems approach for avoiding common failure modes, and loss of sufficient confidence in maintaining CoK that re-verification is required.

When a monitoring system is selected properly, there will be sufficient confidence that knowledge of the nuclear material was attained, and that CoK was maintained for drawing an overall safeguards conclusion—with confidence.

## 4. Safeguards CoK Scenarios

Safeguards activities can be grouped, at a high level, into three categories: stable, dynamic, and transportation. Stable facilities are those where nuclear material is typically stored and safeguarded in place after arrival, or moved on a slow, predictable schedule (i.e., scheduled shipments). Dynamic facilities are those where safeguarded nuclear material movement is a constant or near-constant activity, most notably bulk processing facilities. Not only is the material moving, but it may be changing its physical or chemical characteristics as well. Transportation is the category where the nuclear material is not at a safeguarded facility, but is in transit between facilities, and possibly between member states as well. It would also be appropriate to discuss “sub-facilities” or material balance areas as stable or dynamic; for example, the vault at a processing plant may have nuclear material in storage with a long dwell (storage) time, especially as compared to the processing areas of the same plant.

Descriptions are given for a fuel storage facility as a typical example of a stable facility, a bulk processing facility as a dynamic facility, and a general transportation scenario.

### 4.1. CoK of Stored Nuclear Material

A stable scenario is fairly straightforward with respect to what is required to attain and then maintain CoK of the nuclear material. When nuclear material is shipped or transferred to the facility, initial knowledge must be established. Ideally, this is done by NMA on site. An inspector then must be able to verify that the material is then actually stored in the declared location. Various tools, such as unique identification numbers (UID) on the nuclear material containers or items, seals to ensure that those containers remain closed, cameras to verify equipment and container movement, gamma or neutron detectors (depending on the nature of the nuclear material) to maintain confidence that the movement was of the declared material, and others, will provide confidence that CoK was maintained through the placement of the item into storage. Once in storage, CoK must be maintained for that nuclear material, possibly for a period of decades or more. Periodic verification of UIDs, tamper indication status of seals, radiation signals, and camera footage can be used to verify that the nuclear material is still in place.

### 4.2. CoK During Bulk Processing

The crucial distinction, in terms of maintaining CoK, between a stable facility and a dynamic one is that in a dynamic facility there is a potential for a loss of CoK *simply due to the bulk processing* itself. To explain this, consider measurement uncertainty and a simple case of moving nuclear material from one container and splitting it among three others via process equipment. The high confidence that all of the reported material is in the first container is replaced by the confidence that all of the material is now in the three secondary containers. Except that it is not all there – some of it is still in the first container as a heel, some of it is in the process equipment as holdup, or some may be in filters. Even if it were possible to easily measure each of these amounts, the total measurement uncertainty will have increased. It is even more important, therefore, to assure that once CoK has been attained, it is maintained throughout the dynamic processes, and it must be maintained at the level of the entire facility, not just for individual items of nuclear material.

As with the stable facility, the nuclear material is, ideally, measured on-site to establish knowledge of the nuclear material quantity. The material will arrive as items or in containers, and thus UIDs and seals can be used to maintain CoK as the material is brought into the facility and prepared for processing (there may be storage areas to be considered, similar to larger stable facilities). Again, cameras, radiation detectors, etc. can be used to verify the movements until the materials enter the process. Once in the process, the physical and chemical nature of the material will be changed. It may be chopped, dissolved, chemically converted, mixed, ground, or any other number of steps until the final product is complete. Sufficient monitoring of the processes is vital to provide confidence that the process is operating as declared; in other words, CoK of the process itself must be attained and maintained. This will require process specific monitoring technologies. For example, a CoK of chemical processing plant for spent fuel would use tank level indicators, liquid flow rate monitors, and pH meters, whereas CoK for an enrichment plant would use load cells and enrichment meters. In either case, with maintained CoK, the desired outcome is sufficient confidence that knowledge of the plant operation can be used to draw a safeguards conclusion.

### 4.3. CoK During Transportation

Maintaining CoK during transportation currently relies upon an active seal attached to a container prior to shipment and verified upon receipt [8]. The goal of the seal is to record all openings and closings of the transport container (e.g., trailer, shipping container, etc.) that occur during transportation. This approach requires an inspector to be present at either the point of shipment *and* receipt or, minimally, at the point of receipt when the seal is removed. The inspector must verify that the seal was properly applied (e.g., through the hasp) and that the collected information accurately indicates that the item was properly received. This verification also may include a confirmation of physical attribute(s) (i.e., weight and/or NDA).

Many shippers prefer (for a variety of reasons) not to provide active monitoring or surveillance during transport; this type of surveillance has been considered too sensitive or intrusive for shippers to consider. However, emerging needs to improve detection of potential diversion during transport (e.g., of spent fuel)warrant greater monitoring during transport. Technologies and approaches that can provide high assurance that fuel cycle materials leaving one location arrive (intact) to the designated location will need to be developed as the number of global fuel cycle activities that require transportation continue to expand.

### 4.4 Confidence that CoK Has Been Maintained

When all of the components used for CoK function correctly, an inspector will have sufficient confidence that CoK has been attained and subsequently maintained to use that knowledge for drawing a safeguards conclusion. When CoK is properly implemented with a systems approach, even the loss of a limited number of components will not reduce this confidence enough to lose assurance that the nuclear material is as declared. There will be many commonalities over the three scenarios in the selection of technology and activities to maintain high confidence that CoK has been maintained.

The potential loss of CoK leads to a “Schrödinger’s cat” thought experiment for safeguards. As in the famous example used to explain quantum probabilities, a safeguards inspector has good knowledge of the state of the nuclear material at the time it is verified (i.e., observed). After that, however, the choice of TID, cameras, inspection frequency, unattended monitoring, process monitoring for bulk facilities, etc. gives (or does not give) confidence that the inspector’s knowledge represents the true state of that nuclear material. The purpose for maintaining CoK is to produce an outcome where the confidence is high enough to base safeguards conclusions on that knowledge.

What is needed is a rigorous analysis of the impact of each type of technology and inspection activity on the confidence that CoK has been maintained. Simple cases, such as depicted in Figure 3, need to be extended to incorporate disparate technologies and activities. The analyses must include an understanding of the causes and impacts of events including (but in no way limited to):

- Potential loss of CoK. For example, due to power failure, communications loss, adversary nuisance tampering, or adversary malicious attack.
- Changes to vulnerabilities due to technology selections. For example, rapid self sampling (like for an electronic active seal) requires a power supply. The need for self-testing to detect tamper creates a potential vulnerability. A requirement that the self-testing be reported to an inspector requires communications, which also creates a potential vulnerability.
- Operational considerations. For example, it is expensive to wire/ install communications cables throughout a plant, so it may be desirable to use wireless signals, which has different potential vulnerabilities than cables.
- Combining disparate data. For example, if a seal appears to be broken, but the camera in the same area shows no undeclared activities, what is the impact on overall confidence in CoK?

## 5. Conclusions

As additional safeguarded facilities, and additional types of facilities, are built and operated, the importance of attaining and subsequently maintaining CoK will continue to increase. Depending on the facility type and use, dependence on CoK could potentially become the primary method by which safeguards inspectors draw a conclusion. To maintain CoK, the safeguards equipment and practices used will need to monitor, or sample, the status of the nuclear material and the CoK components at a rate sufficient to ensure that potential tamper and diversion pathways will be detected. The sampling rate will also need to be appropriately nested to account for varying monitoring needs such as electronic self-monitoring vs. visual inspection. Selection of technology and actions to maintain CoK at safeguarded facilities needs to take into account the declared activities, the nature of the facility, and potential diversion pathways. Selected components to maintain CoK must be considered together as there may be interdependencies of failure modes and each technology has its own potential vulnerabilities. A rigorous

analysis of the confidence that CoK has been maintained is needed to assist safeguards inspectors in understanding when they have sufficient confidence that CoK has been maintained to draw safeguards conclusions.

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# Measurement of electron betatron beam parameters

**Michael Kröning, Andrey Lider, Maxim Rychkov, Alexandr Shestak, Dmitry Demyanyuk, Roman Laas**

National Research Tomsk Polytechnic University
E-mail: roman.laas@gmail.com, shap@tpu.ru

***Abstract:***

*Electron beams of mid-range energy between 1 MeV and 10 MeV and their Bremsstrahlung are on increasing industrial demand. Compared with linear accelerators, Betatron accelerators have the advantage of mobile low-cost sources that can be operated by trained technicians according to X-ray inspection standards, for example. Therefore, Betatron accelerators are used for a variety of high energy X-ray inspection and control applications suc h as material and technical structure inspection and cargo control. In response, we continue with further improvements of the Betatron technology. Betatron performance depends on source characteristics resulting from electron beam parameters. For better understanding of design improvements we have to monitor the electron beam parameters during the whole acceleration cycle inside of the Betatron torus. There are various established methods for electron beam characterization. However, we may only apply those methods that allow the beam analysis through the torus wall. As an optical method a high-speed camera is used for monitoring cross-section variations in time. This technique informs about beam distribution during the whole acceleration process as input data for calculating important Betatron parameters like intensity of emitted radiation, beam angle, and resulting effective resolution. The circulating electrons ionize the residual gas in the torus ring, an observable effect due to the relatively high electr on current. Placing signal electrodes below and above the electron beam inside the vacuum chamber we create a potencial difference, that allows us to collect secondary electrons and ions to measure ionization current. We accomplished the design for the later instrumentation by first supporting experiments. Applied to Betatron accelerators, the instrumentation will help to increase the efficiency of future systems applied for material and cargo inspections.*

**Keywords:** electron beams, betatron, monitoring, security

## 1. Introduction

Electron beams of mid-range energy between 1 MeV and 10 MeV and their Bremsstrahlung are on increasing industrial demand. Compared with linear accelerators, Betatron accelerators have the advantage of mobile low-cost sources that can be operated by trained technicians according to X-ray inspection standards, for example. Therefore, Betatron accelerators are used for a variety of high energy X-ray inspection and control applications such as material and technical structure inspection and cargo control.

In response, we continue with further improvements of the Betatron technology. Betatron performance depends on source characteristics resulting from electron beam parameters. For better understanding of design improvements we have to monitor the electron beam parameters during the whole acceleration cycle inside of the Betatron torus. Until today we characterize the beam parameters during electron injection and after ejection only. One of the most important parameters is line profile of beam, which is current density-distribution function in a cross section. The maximal value of integral of this function is in proportion to beam current. Characteristics of density-distribution of current make it possible to identify beam fluctuation type and find out ways to stabilize it. The measurement of a beam parameters needs a special afford in regard to betatron.

## 2. Ionization method

Ionization method is transparent and can be applied to measure both current and beam center continuously. It is based on phenomena of residual gas ionization by electrons. The vacuum in camera is about $10^{-6}$ mm of mercury column, the number of gas molecules is big enough. The ionizing ability of electrons is quite low, much lower than ability of protons, but circulated current is much bigger, than in proton accelerators, because of small mass and big velocities. The suggested system includes sensors of intensity, location and density distribution of the beam. The principal scheme of measures is showed on figure 1.

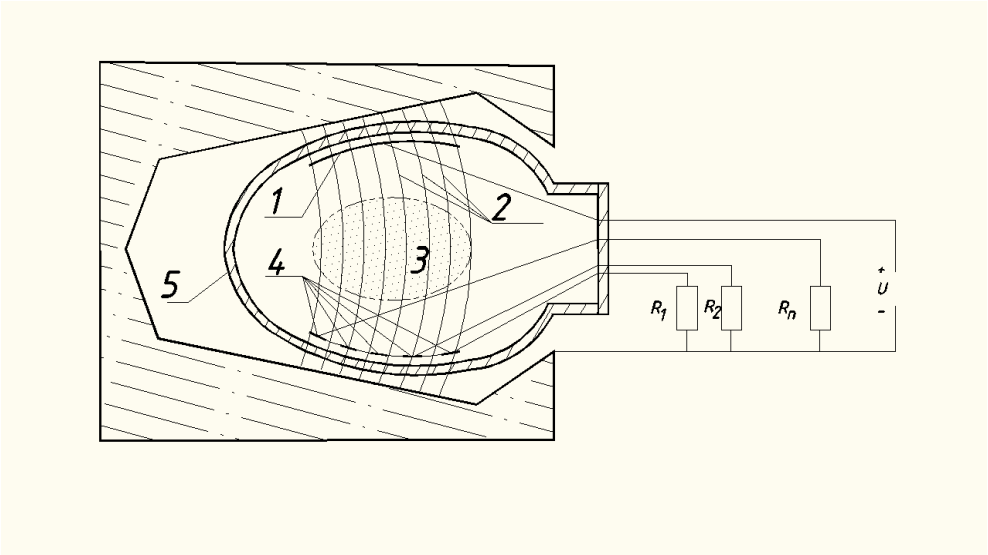


Figure 1: Construction of ionization detector

The number of gas ions and secondary electrons is in direct proportion to current. When the current of circulated beam is 5A, the number of secondary electrons is about $10^{14}$ for each cm. The electrodes 1 and 4 are placed above and below the beam 3 inside the torus. The negative electrode is under voltage U created by high-voltage generator. The positive collector-electrode 4 is divided into 10 parts, each part is ground through resistance $R_n$, where n is the number of collector section. Secondary electrons are moving along the magnetic lines(2) of control field and through the resistances flow to the ground. Voltage on each resistance is in proportion to current of accelerated electrons above each section. The sum of currents is in proportion to the whole beam current. As a guard electrode the conductive coating inside the acceleration camera is used.

The basic characteristic of this method is its sensibility, which can be defined as ionization current $I_i$, taken from collector divided through circuit current Ic of the whole beam.

$$S_I = \frac{I_k}{I_c} = klpB(E) \tag{1}$$

Where k – coefficient of secondary electrons collection, l – length of a section of collector, p – residual gas pressure, B(E) – possibility of ionization, that is defined as the number of pairs of ions, created on 1 cm of path under the pressure 1 Mpa by 1 proton or electron.

$$B(E) = \frac{(\frac{dE}{dx})_{ion}}{\varepsilon p_0} \tag{2}$$

Where p0 – atmosphere pressure, ε – average energy needed to create 1 pair of ions, $(\frac{dE}{dx})_{ion}$ – ionization loss – average quantity of energy lost by particle on ionization on 1 cm path.

The dependence of sensibility on accelerated particles energy is defined by B(E), and the behavior is defined with ionization loss. For electrons this loss can be defined as

$$-(\frac{dE}{dx})_{ion} = 4\pi e^4 N_a Z_a \sqrt{e/2} \frac{\ln(mv^2/2I)}{mv^2} \tag{3}$$

and for relativistic particle

$$-\left(\frac{dE}{dx}\right)_{ion} = 2\pi e^4 N_a Z_a \ln\left[\frac{\left(E^5/2mc^2I^2\right)+1/8}{mc^2}\right] \quad (4)$$

where $N_a$ - number of atoms in 1 cm$^2$, $Z_a$ - atomic number of a substance, I – average ionization potential.

According to (3) and (4) ionization loss and sensibility of described method are in proportion to $1/\beta^2$ in area of low energies ($\beta<<1$, where $\beta$ is relative speed of electrons). The minimum of function is when energy is $2{,}5m_0c^2$, than sensibility growth with lnE. Protons and electrons of the same velocity have quite similar losses of energy on ionization.

To measure the position of balance center, collector electrode is divided into 2 parts diagonally. Differential signal is measured. It is in proportion to radial displacement of a beam, but it also depends on the intensity of a beam. To avoid this dependence the total signal, that also depends on intensity of a beam is measured. Differential signal is divided through total.

$$I_c = Ql\beta c/2\pi r_0 \quad (5)$$

where Q – total circulated charge, c – speed of sound, r0 – radius beam orbit. Than

$$\Delta U = (I_1 - I_2)R = RQl\beta ckp\Delta rB(E)/2\pi r_0 m \quad (6)$$

$$\Sigma U = (I_1 + I_2)R = RQl\beta ckp\Delta rB(E)/\pi r_0 m \quad (7)$$

As we can see from (6) and (7) ΔU/ΣU does not depend on Q or B(E)

## 3. Optical method

For diagnosis we chose optical method: The glowing of residual gas resulted by interaction with electrons was detected.[1, 2]. This method is noninvasive (does not affect beam) and effective for electron beams with relatively low energy. The gas in torus consists mostly of Nitrogen, for it is the main component of air. The most convenient line to measure is 391,4 nm excited $N_2$ molecule line. The cross section of the process has maximum value $14{,}8*10^{-18}$ cm$^2$ for energy of electrons near 100 eV.[3]

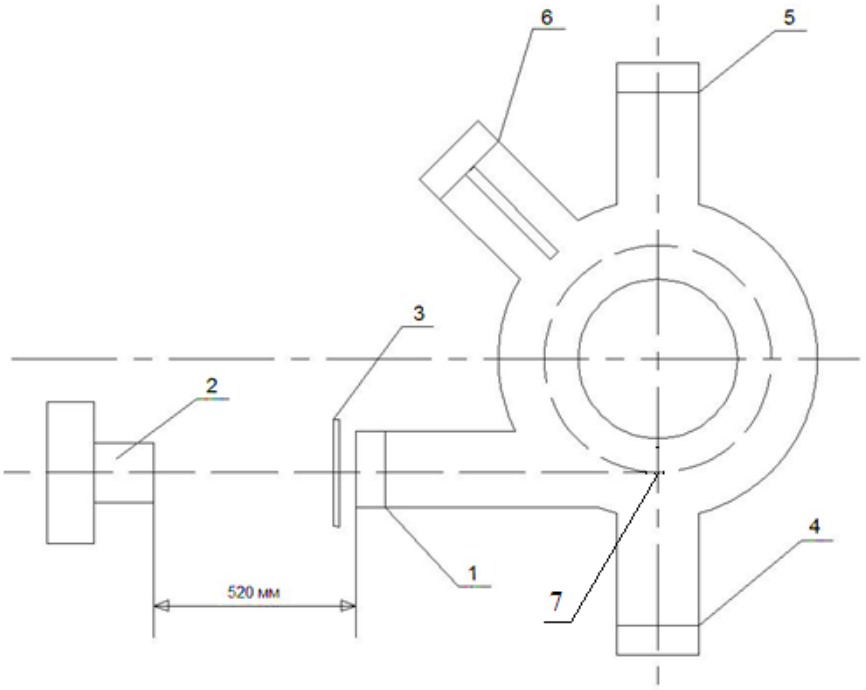


Figure 2: Scheme of optical system: 1-optical window, 2-camera, 3-light filter, 4-free nozzle, 5-vacuum system nozzle, 6-injector nozzle, 7-focal point.

The betatron MIB-4 was used to conduct the experiment. The scheme is shown on figure 2. Residual gas pressure during the experiment was $8 \times 10^{-6}$ , injection current – 0.7A, acceleration impulse frequency – 400Hz. The first experiments were trial; that is why results are not gauged by magnetic system of the betatron. However, matrix of the camera was calibrated. Initial data was intensity of light dependence on pixel number. With this geometry, calibration coefficient appeared to be 0.02 mm/pixel.

In order to identify frequency content of residual gas glowing we used different color filters. The presence of filters in system also decreased the background light, which was emitted by hot electrode in red region.

The maximum level of glowing appeared to go through filter CC15. It`s pass band is shown on figure 3a.

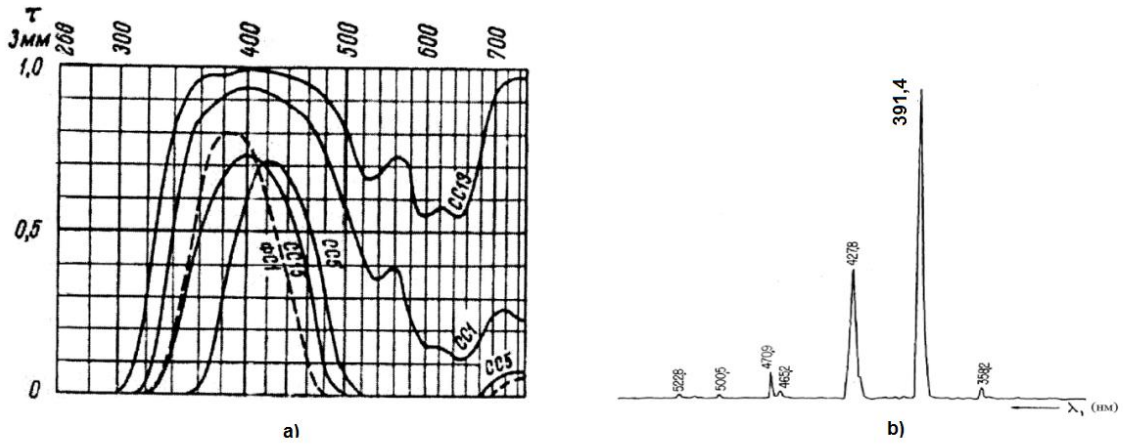


Figure 3: a) Light filters transmission curves; b) Nitrogen glowing spectrum.

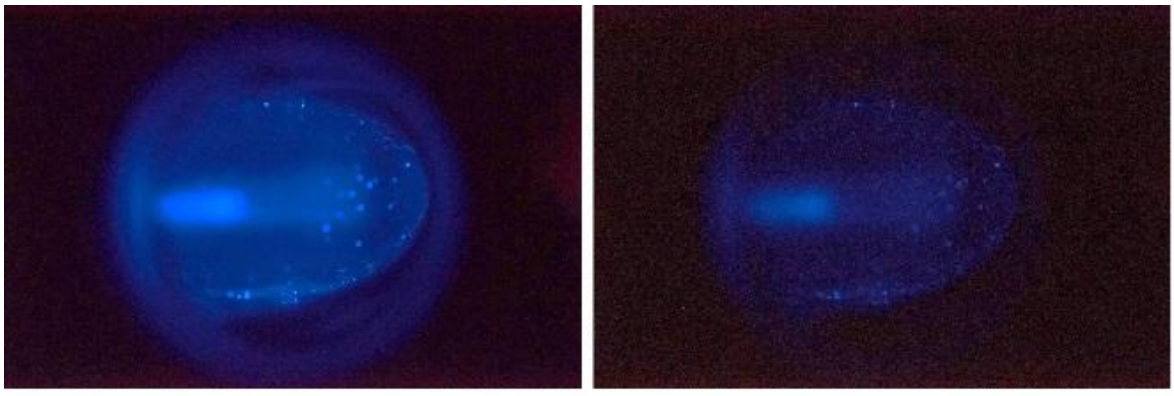


Figure 4: a) Exposure time 1 second; b) Exposure time 0,25 seconds.

Figure 3b shows spectrum of Nitrogen[4]. The most intensive lines 391,4 nm and 427,8 nm lay in passband of the filter. During the experiment, important data on sensitivity of the method was derived. Betatron has relatively small current inside torus, the sensitivity of method and equipment has to be high enough. To estimate it, we used different exposure time. With the exposure time more than 3 seconds matrix appears to be oversaturated, so that some data is lost. The shortest exposure time that we set was 0.25 seconds. Ordinary DSLR camera with uncooled matrix made it impossible to decrease the time, because noise level appeared to be very high.

Figure 5a shows distribution of current density in horizontal crossection. Right sight of the peak can be easily approximated by Gaussian distribution with mean square deviations $\sigma = 2{,}19$, and FWHM=5,16. Deviations on the left side can be easily explained: the picture shows us not the real cross section, but all the light that camera can capture (figure 2).

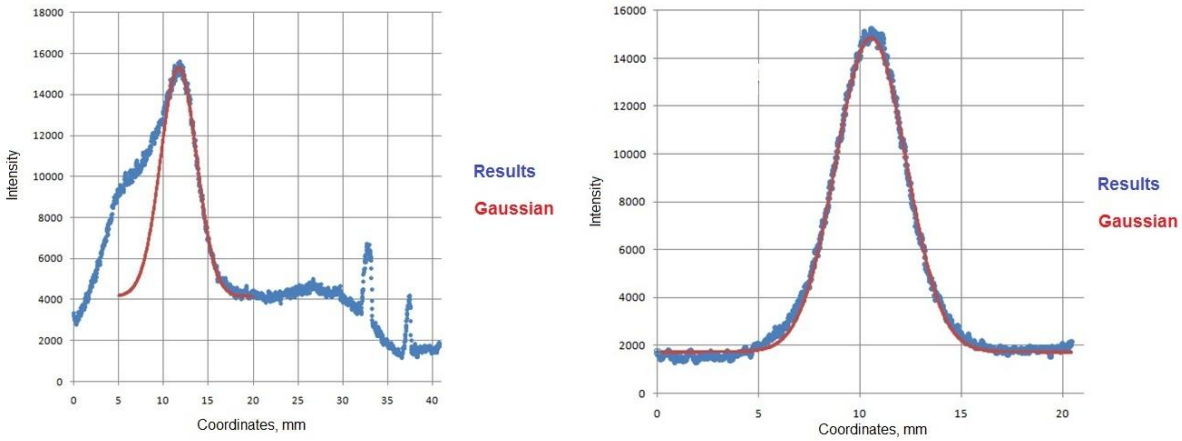


Figure 4: a) horizontal distribution graph; b) vertical distribution graph

Figure 5b is distribution of current density in vertical cross section. It can be approximated by Gaussian distribution with mean square deviations σ = 2,19, and FWHM=5,16. Deviations on the left side can be easily explained: the picture shows us not the real cross section, but all the light that camera can capture (figure 2).

So that the application of the method gave us figures of integral current density distribution pro 100 acceleration impulses with energy 4 MeV.

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# Preliminary Next Generation Safeguards Initiative Concept of Operations for a Cylinder-Monitoring System

**J. Michael Whitaker, Jessica White-Horton, Oak Ridge National Laboratory**
**Karyn Durbin, National Nuclear Security Administration**

E-mail: whitakerjm@ornl.gov, karyn.Durbin@nnsa.doe.gov, Whitejl@ornl.gov

***Abstract:***

Because enrichment capabilities are expanding and black markets exist for enrichment technology, the content of natural and low-enriched uranium hexafluoride ($UF_6$) cylinders used in the front end of the commercial nuclear fuel cycle can appear to be more attractive to a potential proliferant. The National Nuclear Security Administration is sponsoring the development of a concept of operations for a global $UF_6$ cylinder monitoring system for the most commonly used $UF_6$ cylinders. The preliminary concept has three key components: (1) a truly unique cylinder identification method that can be standardized across the industry for use by both industry and inspectorates; (2) a global cylinder "registry"; and (3) an unattended, on-site monitoring system to ensure that only declared, registered cylinders are processed at key operational nodes. The strategic objectives are to (1) significantly shorten the time it takes to reconcile shipments between countries and to detect the diversion of a $UF_6$ cylinder and (2) enhance the International Atomic Energy Agency's capability to detect the introduction of undeclared cylinders and the misuse of declared cylinders at safeguarded facilities. This paper provides the functional requirements for the cylinder UID, the necessary inputs for the global registry, and a preliminary concept for how facilities and inspectorates would interact with the registry and the unattended monitoring systems.

**Keywords: standardization, $UF_6$, cylinders, nonproliferation, integrated monitoring**

## 1. Introduction

With growing nuclear commerce and new safeguards initiatives to achieve the IAEA's verification objectives for States as a whole, the international safeguards system continues to face ever-increasing demands to verify nuclear materials and detect clandestine activities. Emerging proliferation threats from both State and non-State actors are also leading to increased concerns over the diversion of nuclear materials - for example, the loss of cylinders containing uranium hexafluoride ($UF_6$) during transport or the introduction of undeclared feed into safeguarded enrichment plants. While the IAEA timeliness goal for detecting the diversion of natural or low-enriched uranium is one year, a proliferant State with access to enrichment technology, with a single cylinder of $UF_6$, could produce a significant quantity (SQ)[1] of highly enriched uranium (HEU) in less than a year. Given a worst case scenario, this could be accomplished in as little as 30-90 days using a small-to-moderately sized enrichment facility. The enrichment could theoretically be performed at either a clandestine facility or by misusing a portion of a safeguarded facility.

## 2. NNSA Global Cylinder Identification and Monitoring Project

To address this nonproliferation concern, the NNSA initiated a 5 year program in 2011 to demonstrate, at the proof-of-concept level, a $UF_6$ cylinder identification and monitoring system.[2] This program builds upon a prior exploratory study sponsored by NNSA, and also upon work by several industry leaders on the merits of unique identifiers for $UF_6$ cylinders.[3,4] The primary objective of this NNSA program is to develop and demonstrate a concept for a global cylinder identification and monitoring system that

could substantially increase the effectiveness and efficiency of the IAEA to detect credible diversion and undeclared production pathways.

The first task in the program involved documenting industry practices relevant to the design of the system, identifying potential cylinder diversion scenarios and undeclared production pathways, and evaluating current practices for detecting the pathways.[5] These HEU production pathways fall into three areas:

- diverting an entire declared cylinder from a safeguarded facility to a clandestine facility,
- misusing a declared cylinder at a safeguarded facility, and
- introducing undeclared material into a safeguarded facility (via an undeclared cylinder).

The second program task was to develop a preliminary concept of operations for a cylinder identification and monitoring system. The use of a unique identifier (UID) for each cylinder is a primary component of the concept. The following recommendations related to this UID were compiled during task 1:

- The UID must be designed to function in all the relevant operating conditions including the wide temperature range and exposure to hydrogen fluoride gas and have a minimum life expectancy of 10 years (currently many cylinders remain in circulation 30-40 years).

- The UID must be tamper indicating, authenticable, and have the capability to be automatically scanned or read in some fashion. This will eliminate the potential for transposition errors and provide for the development of unattended monitoring systems.

- For new cylinders, the UID would be installed during fabrication. Until the fabrication standards are revised, UIDs for new 48Ys could be applied at conversion plants, and UIDs for 30Bs could be applied at enrichment plants.

- For existing cylinders currently in active circulation, the UID could be applied either during the recertification process or when they are routed back through conversion or enrichment plants to be refilled.

Additional observations related to the overall system included:

- Cylinders containing more than an SQ that are placed in long-term static storage should be randomly checked more frequently than once a year.

- A complete “registry” of all $UF_6$ cylinders, coupled with a capability to read the unique identifier on “registered” cylinders at carefully selected nodes in a safeguarded facility, could significantly increase the capability of quickly recognizing the presence of any unregistered cylinders. A registry of cylinders would also improve the IAEA capability to match transfers between countries.

- Additional safeguards measures for continuously monitoring throughput at conversion and enrichment facilities are needed to address the undeclared production scenarios. A cylinder identification and monitoring system could be a key enabler of a robust safeguards approach that incorporates such additional measures.

- Should a cylinder be diverted from a site or during transport, some form of cylinder monitoring or surveillance would prove helpful by establishing the last known location of the cylinder.

- Cylinder monitoring would also prove valuable in detecting scenarios involving content swapping or nameplate switching.

- Monitoring of empty cylinders would prove valuable in detecting scenarios involving feeding undeclared material using such cylinders.

## 3. Preliminary Concept of Operations

Based on the analysis in Task 1, the team developed a preliminary concept that includes four primary components:

- a standardized, remotely readable, tamper-indicating unique identifier (UID) to be applied to each cylinder
- UID readers (portable and installed) at cylinder-handling facilities
- Access-controlled, global cylinder "registry" (or database)
- IAEA safeguards cylinder monitoring system (for use at safeguarded facilities)

The integration of these components is illustrated in Figure 1.

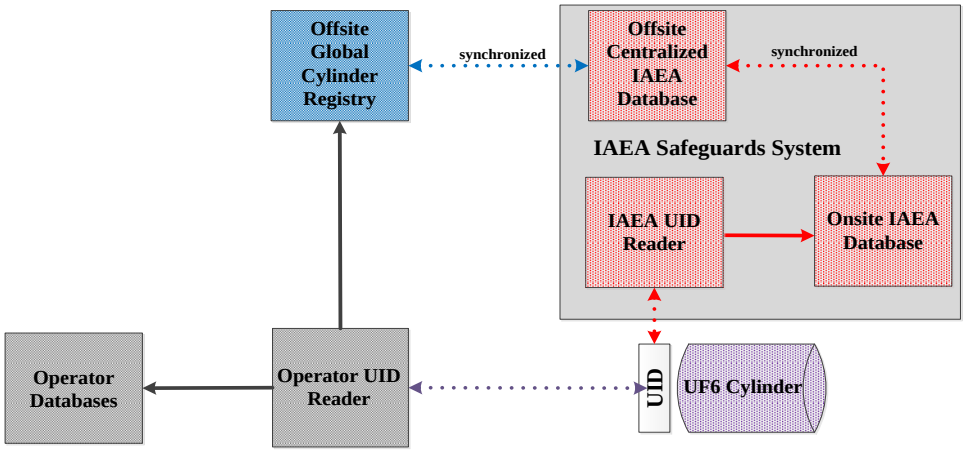


**Figure 1. Fundamental components of conceptual global cylinder identification and monitoring system**

It is important to note that for the purposes of this project, the team is defining "***monitoring***" as: "*a practice that provides the location and status of all $UF_6$ cylinders in the commercial nuclear fuel cycle."* Monitoring in this case is distinguished from "***tracking***" which the team is defining as: "*actively following the path or trail of a cylinder as it moves from location to location."* The team does not currently envision *tracking* $UF_6$ cylinders as part of the concept.

The concept of operation that is being developed is for a "monitoring" system that maintains information on the location and status of each registered cylinder.

### 3.1. Unique Identifier (UID)

The UID is the cornerstone of the identification monitoring system. Each cylinder would have a UID that could be used for identification and monitoring and would be designed for use across the entire $UF_6$ cylinder industry. A fundamental feature of the UID is that it can be used independently by both the plant operators and the IAEA. High level functional requirements for the UIDs include:

- Standardized format/design
- Truly unique across all industry
- Tamper-indicating for use by IAEA for safeguards purposes
- Contains necessary characteristics to provide for authentication
- Capability to be remotely read
- Would ***not*** replace the current cylinder nameplate, but can be correlated with the identification number stamped on the nameplate
- Useable by both the operator and the IAEA
- Ability to withstand environmental conditions and cylinder handling practices
- Capability to be applied at either the cylinder fabricator or in the field
- Minimum life expectancy of 10-30 years

A standardized UID would be attached to all 30B and 48Y cylinders. The use of a standardized UID provides many direct benefits to the international safeguards regime and to industry. A standardized cylinder ID could produce immediate improvements in reporting to SSACs and the IAEA by reducing reading and transcription errors.[6,7,8] The industry's use of a standardized format would eliminate the need for companies to apply additional label and identification numbers to the cylinders that adds confusion to various reporting systems. Eliminating multiple identifiers will directly benefit the IAEA in reconciling cylinder transfers between facilities and countries and in verifying facility records during on-site inspections.

The UID would be tamper indicating and equipped with a characteristic that provides for "authentication." Having a tamper-indicating UID that is acceptable to the IAEA for safeguards purposes could improve the efficiency of the IAEA in verifying cylinder IDs in the cylinder yards. It can be time consuming accessing and trying to read the tiny ID number stamped on the nameplates at the end of the cylinder. Applying the UID at a more observable location would streamline this routine, repetitive inspection activity. Having an authentic, tamper-indicating UID would also give the inspector increased confidence that the cylinder identifier had not been falsified in an effort to conceal a diversion or facility misuse, thereby enhancing the effectiveness of IAEA safeguards.

Designing the UID to be read remotely would substantially reduce the time required for the IAEA to verify cylinder inventories during on-site inspections (provided that the IAEA can use their own portable UID readers during inspections). This quicker cylinder inventorying capability would provide for increasing the overall efficiency with two important benefits: 1) during normal inspections, inspectors would be able locate and verify declared cylinders much quicker, leaving more time for activities to detect undeclared production pathways (e.g., design verification, looking for undeclared cylinders and operations); and 2) the IAEA could verify "static" cylinder inventories more frequently than once a year during PIVs if such increased frequency was deemed necessary. Portions of the static inventories could be verified on a random basis during scheduled or unannounced inspections to address the diversion and undeclared production pathways. Additionally, this would reduce the radiation exposure of inspectors, especially in the areas containing emptied cylinders.

### 3.2. Unique Identifier Readers

The UID readers could either be hand-held systems or installed, unattended systems. Operators, regulators, and the IAEA inspectors could have independent capabilities for reading the cylinder UIDs. Unattended readers could be installed at strategic locations that could benefit operators and/or IAEA (e.g., cylinder receipt & dispatch areas, accountability scales, feed stations, etc.). For unattended readers, the information for reading cylinder UIDs would likely need to be integrated with a cylinder database to verify that a read UID is in fact present in the database. Cylinders read for the first time (and not in the database) would need to be "registered" (i.e., issue/record the ID).

The operator would have the capability to read the cylinder UID and enter information into facility databases that monitor information related to the location, status and utilization of cylinders. Any negotiated IAEA system could collect information through the use of unattended readers, as well as hand-held readers that may be utilized by inspectors during on-site inspections. Information to be submitted to a global registry could be collected either manually, with hand-held readers or with unattended reader systems.

### 3.3. Access-controlled, Global Cylinder Registry

An access-controlled global registry would be established for all $UF_6$ cylinders (beginning with 30B and 48Y in active circulation) in both NWS and NNWS. Cylinder-related information captured in the registry would, at a minimum, include:

- UID
- Facility location
- Date
- Cylinder owner
- Status of cylinder (full, empty, material type)

The registry could receive inputs from the operators, national authorities, IAEA, and from other existing sources:

- Fabrication of new cylinders
- Destruction of old or damaged cylinders
- Shipments of empty cylinders
- Cylinders containing $UF_6$ at IAEA safeguarded facilities
- Shipments of $UF_6$ between IAEA safeguarded facilities
- Cylinder inventories in NWS

For IAEA-safeguarded facilities, the safeguards-related information in the registry could be periodically synchronized with the IAEA safeguards system. This global registry, combined with current IAEA safeguards activities, could provide for more timely detection of diversion and undeclared production of enriched $UF_6$. A global registry would give the IAEA the ability to generate a list of all cylinders in the State which would support State evaluations, annual inspection planning, and acquisition path analyses. A registry would be especially valuable in IAEA efforts to reconcile cylinder transfers (i.e., transit matching) in a more timely manner.

For IAEA safeguarded facilities, the registry could provide the IAEA with a technical basis for recognizing unregistered (and potentially undeclared) cylinders at a facility. Depending on how frequently information is submitted to the registry and the IAEA safeguards department access to the registry, it is possible that the inspectors could have access to cylinder receipt information prior to inspecting facilities. This information would be valuable in planning inspections (especially unannounced or short-notice random) and verifying operator receipts records.

If the IAEA uses State Level Concepts to differentiate verification methods applied at a specific enrichment plant in a specific country, a State implementing cylinder monitoring and transparency in cylinder movements could be a state factor signaling to the IAEA that a reduction in traditional HSP safeguards measures may be reasonable.

### 3.4. IAEA Safeguards Systems (for facilities safeguarded by the IAEA)

The IAEA safeguards system utilizing UID information would likely consist of on-site databases (containing analysis algorithms) at each facility and an off-site centralized database. UID information would be inputted into the on-site IAEA system from unattended UID readers and hand held readers utilized by inspectors during the on-site inspections. The synchronization of information between the on-site databases and the centralized database would be evaluated and designed to address the detection times for the diversion and misuse scenarios.

A capability for the IAEA to install unattended readers at key operational nodes would significantly increase their ability to assure the absence of undeclared nuclear activities within a State by being able to verify that only declared cylinders are being processed. Once proved reliable, this continuous monitoring of flow could lead to reductions in other inspection activities. Monitoring high traffic areas like the feed stations would allow for the detection of cylinders leaving the facility that have not had their contents fed (and therefore may be diverted for clandestine production).

### 3.5. Strategy for Phased Implementation

Initially the project will only focus on model 30B and 48Y cylinders in active circulation between sites. The UID would be attached to each 30B and 48Y cylinder. While it is optimal that all cylinders purchasers use the same format/design for the identification number, it is recognized that many companies have invested resources for the format that they are currently using. Thus, initially, a limited number of well-defined identification formats may be required as the program is phased-in over time.

Ultimately it would be desirable for the UID to be applied during the initial fabrication of the cylinder – this will likely require a revision to the international cylinder standards (ANSI & ISO). During a phased-in transition period, a capability to apply the UID to cylinders already in circulation would be required. It

is anticipated that it will take 3-5 years to apply the UIDs. Most likely, the UIDs could be applied as part of the cylinder certification process.

## 4. Initial Feedback from Stakeholders

Representatives from industry have been engaged throughout the project through international conferences, topical technical meetings, and individual briefings and discussions. In early 2013, this preliminary concept of operations was briefed to representatives of industry, national authorities, and inspectorates that could be affected by its implementation. The initial feedback was that all companies seemed interested in an industry-wide UID for cylinders. There were questions regarding the cost, who would apply, and what happens if destroyed or damaged. Some companies have or are looking at automated reading systems for the company labels that they create and apply to cylinders at their individual facilities. Although some companies and regulators have registries for cylinders under their ownership or jurisdiction, there is concern over what information would be contained in the global registry & who would have access.

The NNSA project team has consistently received valuable feedback from candid discussions with cylinder-handling stakeholders throughout the project. This feedback is being incorporated into the concept of operations as it evolves.

## 5. Next Steps

In defining the baseline requirements and developing the preliminary concept of operations, the project team has been careful to not "jump to a solution" by pre-selecting a particular UID technology (e.g., barcode, radio-frequency identification device, etc.). The next project steps include investigating available technologies and down selecting options. This process will consider, among other factors, the total life cycle costs. More work needs to be performed on designing the architecture and operation of the global registry. A phased-implementation strategy needs to be developed and discussed. Key components and operational aspects need to be tested and evaluated in operational settings. Throughout these next steps, engagement with key stakeholders will continue to be a priority to ensure that both concerns and desires are identified and integrated into the design.

***Acknowledgement***

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# Determination of the half-life and specific thermal power of $^{241}$Pu by nuclear calorimetry

Stephen Croft[1], Peter A. Santi[2], and Robert D. McElroy, Jr.[1]

1. Safeguards & Security Technology, Global Nuclear Security Technology Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA.
2. Nuclear Engineering and Nonproliferation Group NEN1, Los Alamos National Laboratory, Los Alamos, New Mexico, USA.

Abstract

At about 14.3 years, $^{241}$Pu has the shortest half-life of the abundant plutonium isotopes in reprocessed irradiated nuclear fuel. Because the half-life and associated uncertainty of $^{241}$Pu affect both the estimation by decay-correction calculation of the total amount of separated plutonium in storage and the determination of the total plutonium mass by non-destructive assay, it is important to know the half-life of $^{241}$Pu with a higher fractional accuracy than that of other plutonium isotopes. This paper addresses the determination of the $^{241}$Pu half-life by measurement of the thermal power of a sealed plutonium source, ideally initially rich in $^{241}$Pu and chemically stripped of $^{241}$Am, as it evolves in time through the use of nuclear calorimetry. The absolute accuracy of nuclear calorimeters can be ensured over long periods of time (many years) using long-lived nuclear reference materials and/or electrical heat standards. Because of this, one can expect nuclear calorimetry to offer an accurate way to determine the half-life that is comparable in quality and independent of, yet complementary to, other approaches. Temporal analysis of the power-versus-time data also yields an estimate of the specific power of $^{241}$Pu.

After describing the principle of the method and developing the pertinent mathematical expressions, we outline the approach by drawing on some unpublished notes of Kenneth C. Jordan, who carried out such an experiment at the Mound Laboratory more than 40 years ago. Today Jordan's work remains possibly the most significant experiment of its type to the data evaluator, however, objectively assigning confidence to it is problematic because details have never been adequately reported. We conclude that a new, high-accuracy nuclear calorimetry campaign to remeasure the $^{241}$Pu half-life and specific power is needed as a means to confirming that systematic biases are under control, support international safeguards, and improve fundamental metrology and nuclear data—for instance, the mean beta energy emitted by $^{241}$Pu.

Keywords: $^{241}$Pu half-life; $^{241}$Pu specific power; nuclear calorimetry

# Assessing Capabilities of Handheld Raman Spectrometer *FirstDefender RM* for Complementary Access Applications

**Andrey Berlizov[1], Doris Ho Mer Lin[2], Adrian Nicholl[2], Klaus Mayer[2]**

[1]International Atomic Energy Agency
Vienna International Centre
PO Box 100, 1400 Vienna
E-mail: a.berlizov@iaea.org

[2]Institute for Transuranium Elements
Joint Research Centre, European Commission
Postfach 2340, 76125 Karlsruhe, Germany
E-mail: doris.ho-mer-lin@ec.europa.eu, adrian.nicholl@ec.europa.eu, klaus.mayer@ec.europa.eu

***Abstract:***

*The paper presents results of an assessment of the performance, applicability and adaptability of the commercially available handheld Raman spectrometer FirstDefender RM (Thermo Scientific, USA) to complementary access (CA) applications. The assessment was conducted jointly by the IAEA and EC JRC Institute for Transuranium Elements (Karlsruhe, Germany). The primary objects of interest in this evaluation were uranium ore concentrates (UOC), frequently referred to as yellow cake. The initial study was focused on finding optimal conditions for the use of the instrument by exploring different modes of operation, measurement vials and sample preparation approaches. After this, a representative collection of 108 UOC samples, covering a wide range of yellow cake of different types, origin and purity, was used to build up the instrument's library. A subset of the UOC samples was then measured as "unknowns" to test the identification capability of the upgraded instrument. The accuracy of the library matches was shown to be sound for uranyl peroxide, sodium diuranate, ammonium diuranate and uranium trioxide types of the yellow cake, for which the identification results could be delivered within a few seconds to a few minutes. The origin of these materials (i.e. mine or ore concentration facility) could be correctly identified in ~50% of test cases. The $UO_2$, $U_3O_8$, uranyl hydroxide and some calcined yellow cake could not be however analysed with FirstDefender RM, pointing thus to certain limitations of this analysis method. The paper summarizes the outcomes of the assessment, which will be further used for the creation of a specialized safeguards-tailored version of the portable Raman spectrometer.*

**Keywords:** complementary access; instrumentation; identification; Raman spectrometer; uranium ore concentrate

## 1. Introduction

Deterring the proliferation of nuclear weapons requires early detection of any misuse of nuclear material and technology. The IAEA has made great efforts to ensure any diversion of declared nuclear material in declared facilities would be detected. A key challenge facing the IAEA now is to improve the effectiveness in detecting any possible undeclared nuclear material and nuclear activities and thereby enhance the overall deterrence of proliferation. To address this challenge, the IAEA's Department of Safeguards, in co-operation with external entities, is undertaking an upgrade of the equipment kit relevant for Complementary Access (CA) and Design Information Verification (DIV) activities.

This complex task is being achieved by: (i) identifying groups of specific materials that may serve as

signatures of different nuclear fuel cycle activities; (ii) creating a representative repository of respective reference materials; and (iii) exploring capabilities of different on-the-shelf technologies and portable instrumentation vis-à-vis the detection and identification of these materials. Among others, the laser based technologies, e.g. the laser induced breakdown spectroscopy and Raman spectroscopy, are under evaluation. This paper describes a recent assessment of the performance, applicability and adaptability of the commercially available handheld Raman spectrometer *FirstDefender RM* [1], which was conducted jointly by the IAEA's Department of Safeguards and EC JRC Institute for Transuranium Elements (Karlsruhe, Germany).

The results of the evaluation of the *FirstDefender RM* against different chemical warfare agents were reported previously by Matthews et al. [2]. Neat chemical warfare agents in liquid form were identified in nearly 100% of cases, while correct identification of mixtures was observed in 57 out of 66 tests. The capability for rapid and correct identification of toxic industrial chemicals was subsequently demonstrated by the same group using a set of 31 relevant materials [3].

It must be pointed out that the default on-board library provided by the manufacturer along with the *FirstDefender RM* already contains a number of safeguards-relevant chemicals, such as formic and oxalic acids, boron trifluoride, lithium carbonate, hydrazine, tri-butyl-phosphate etc. The evaluation of the identification capability with regard to these chemicals as well as the extension of this list is one of the future tasks highlighted at the end of this paper. As a part of it, the objects of interest in the current evaluation were uranium ore concentrates (UOCs), whose safeguards relevance is defined in the basic safeguards documents.

In particular, pursuant to Articles 34(a) and (b) of INCIRC/153 [4], States are obliged to notify the Agency about imports or exports of this kind of material (frequently referred to as "pre-34c-material"), and therefore a technical capability for the verification of this information is essential. Provision of this technical capability is also important for the verification of a State's declarations under its additional protocol, which obligates the reporting of the locations, quantities, chemical composition, the use or intended use or import/export of any source material, which has not reached the grade suitable for fuel fabrication (Article 2.a.(vi) of INFCIRC/540 [5]). Taking into account these considerations, the capability of the *FirstDefender RM* with regard to the identification of the type and origin of different industrial UOCs was one of the particular topics addressed in the current evaluation.

## 2. Experimental

### 2.1. Raman spectroscopy

The Raman spectroscopy is an active interrogation technique commonly used to observe vibrational modes in different materials and chemical compounds [6]. It relies on inelastic scattering, or Raman scattering, of quasi-monochromatic light from an interrogating laser. The intensity of the process in comparison to the intensity of incident light and elastic Rayleigh scattering is defined by the following ratios: $1 : 10^{-6} : 10^{-10}$ for incident light, Rayleigh scattering and Raman scattering, respectively.

In Raman scattering the laser light interacts with molecular vibrations, resulting in the energy of the incident photons being decreased (the Stokes component) or increased (the anti-Stokes component). This change in energy is called the Raman shift. Each type of molecular bond produces a characteristic Raman shift, and subsequently every chemical compound produces its own pattern of Raman shifts, which can be used for its identification. The Raman spectra represent Raman shifts as function of the wave number. Typically, only the Stokes part of Raman spectrum is analyzed, as under normal conditions it exhibits a much stronger intensity.

The Raman spectroscopy has become increasingly popular, due to its vast applications in the field of military, homeland security, forensics, medicine, material sciences, pharmaceutical sciences etc. Several models of portable Raman spectrometers are currently available in the market, including those manufactured by BWTEK Inc [7], Agiltron [8], Rigaku Raman Technologies [9], Thermo Scientific [10], Analytik Ltd [11].

Normally, a near infrared diode laser in combination with charge-coupled device (CCD) based spectrometer is used in portable Raman instruments. The near infrared diode lasers provide

substantial reduction in fluorescence, have sufficient power, compact size, long lifetime and of low cost. Furthermore, extensive databases of Raman spectra of various chemical compounds have been measured with this type of excitation source and such databases are readily available. However, for certain applications, this type of lasers can be too weak. Another drawback is the elevated noise of CCD that deteriorates Raman response, especially in field operations [12].

## 2.2. Equipment

The *FirstDefender RM* examined in this work is a light-weight (m = 800 g) rugged handheld spectrometer manufactured by Thermo Scientific [1]. The instrument is based on the 785 nm continuous diode laser and a CCD spectrometer covering the spectral range of Raman shifts from 250 to 2875 $cm^{-1}$. The spectral resolution of the spectrometer varies from 7 to 10.5 $cm^{-1}$ across the spectral range. The laser output power is adjustable by 3 levels: low, medium and high, which correspond to 75 mW, 125 mW and 250 mW, respectively.

The *FirstDefender RM* offers two possibilities for measurements (Fig. 1). One is the point-and-shoot geometry, which is a simple and very fast way of analysis of materials contained in translucent enclosures. Another possibility is the in-vial geometry, where a small amount of material is put in a standard 4 mL glass vial and examined inside an integrated vial holder. The instrument does not require a calibration. A quick functionality check of the instrument is periodically done by measuring a reference polystyrene sample.

The instrument's firmware fully automates the spectrum acquisition and analysis. The processing of acquired spectrum includes the removal of fluorescence continuum, followed by smoothing and extraction of the peak related information. The material identification is then performed by matching the extracted information with data entries in the extensive built-in library of Raman responses enhanced by sophisticated chemometric algorithms for the determination of presence of mixed and contaminated chemicals. The built-in library is upgradable such as new compounds can be added using a special procedure for performing library scans.

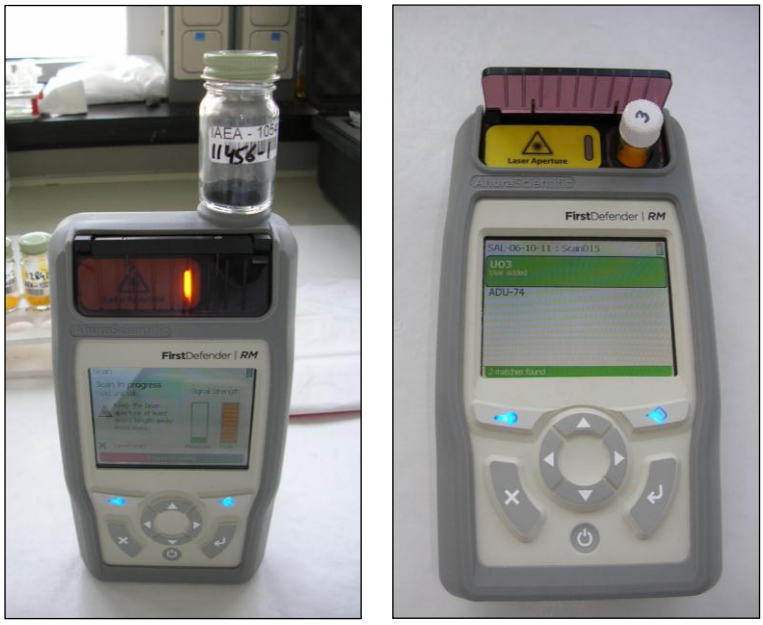


**Figure 1:** The *FirstDefender RM* portable Raman instrument performing analysis of an UOC material in the in-vial (left) and in the point-and-shoot (right) measurement geometries.

## 2.3. Materials

A total of 108 UOC samples were measured in the course of this study. Among these were 75

samples provided by Nexia Solutions, Springfield, UK; these samples originate from a large number of uranium mines and milling facilities around the world. Additionally examined were 19 samples from Blind River and Port Hope, Canada; 5 samples from Beverly and Ranger, Australia; and 9 materials from unspecified locations available as safeguards samples in the Agency's Nuclear Material Laboratory in Seibersdorf. Due to different technological processes involved in the production of these samples, they represented different forms of yellow cake with following major components: uranium peroxide ($UO_4$), uranyl oxide ($UO_3$), tri-uranium octa-oxide ($U_3O_8$), ammonium diuranate (ADU, $(NH_4)_2U_2O_7$), sodium diuranate (SDU, $Na_2U_2O_7$) and uranyl hydroxide ($UO_2(OH)_2$).

In addition, 5 yellow cake samples were synthesized in the laboratory from pure starting materials under well controlled conditions and measured for the sake of comparison. These included uranyl peroxide, ADU, SDU, uranyl hydroxide and also ammonium uranyl carbonate (AUC, $(NH_4)_4UO_2(CO_3)_2$), which was not available as an industrial sample.

For the purpose of measurement conditions optimization, a set of standard 4 mL glass vials were purchased from different manufacturers, namely: PilotVials (12-1 ST), Ziemer (1.300104, 3.300357, 2.300307), Sigma-Aldrich (Z291935) and Perkin Elmer (L9001029).

### 2.4. General approach

The initial study was focused on finding optimal conditions for laboratory and in-field use of the instrument by exploring different measurement vials, the in-vial versus point-and-shoot analysis options, and sample preparation approaches. The results of these studies are presented in Section 3.1.

After this, the UOC samples, covering a wide range of industrial yellow cake of different types, origin and purity, were measured and obtained scans were appended into the instrument's library. The spectral features of the Raman spectra of different UOCs are presented and discussed in Section 3.2. A subset of the UOC samples was then measured as "unknowns" to test the identification capability of the upgraded instrument. The respective results are presented in Section 3.3.

Finally, the *FirstDefender RM* was tested against the uranyl nitrate hexahydrate (UNH) and some other uranium bearing materials. A separate study was conducted to explore the potential application of the instrument for a quantitative determination of the uranium concentration in uranyl nitrate solutions. The results of these studies are not presented in this paper and will be published elsewhere.

## 3. Results and discussion

### 3.1. Optimization

#### 3.1.1. Vials

Although Raman signal of glass is known to be weak and featureless, for a material with feeble Raman response it may cause interference and thus pose difficulties in the interpretation of measurements. Fig. 2 compares Raman spectra from different empty vials acquired for the same fixed time of 10 s. All spectra exhibit a broad bump spanning the interval of wavenumbers from 1200 $cm^{-1}$ to 1800 $cm^{-1}$ with the maximum near 1350 $cm^{-1}$. It does appear that this bumpy distribution is at least two times smaller for vials that were purchased from Aldrich. The latter were therefore selected for future use.

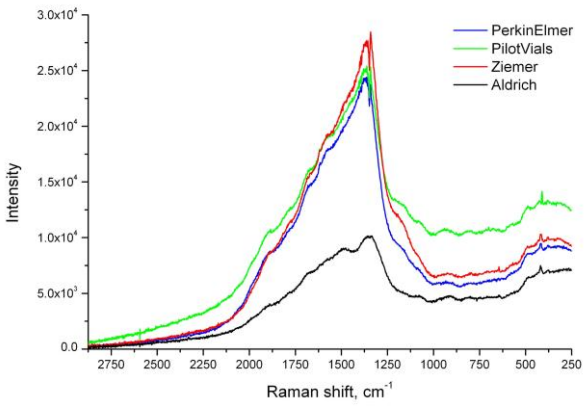


**Figure 2:** Raman spectra from different empty vials.

Fig. 3 and 4 show Raman spectra of ethanol and water. It is seen that, due to the strong molecular signal of ethanol, the differences amongst spectra collected in different vials are subtle. On the other hand, the spectra taken from water, which produces a very weak Raman signal, are essentially affected by scattering in the vial material. For instance, the water peak at 1600 $cm^{-1}$, which is a typical of O-H bend, is poorly recognizable in vials that exhibit larger scattering signal. The use of vials with lower scattering response may therefore also reduce the measurement time for materials with a weak signal. Note that, due to the large difference in the signal to background ratio for ethanol and water, the duration of the automatic spectrum acquisition in both cases differed by a factor of ~200.

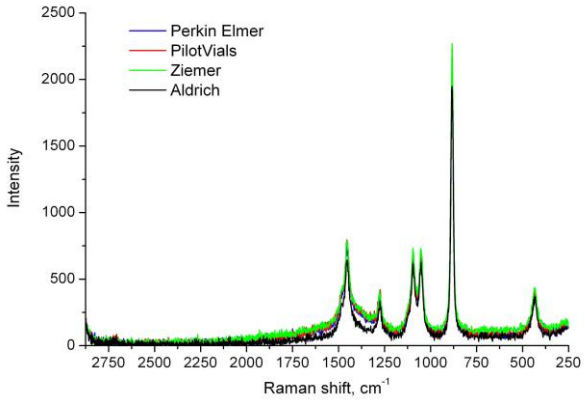


**Figure 3:** Raman spectra of ethanol in different vials (measurement time - 5 s).

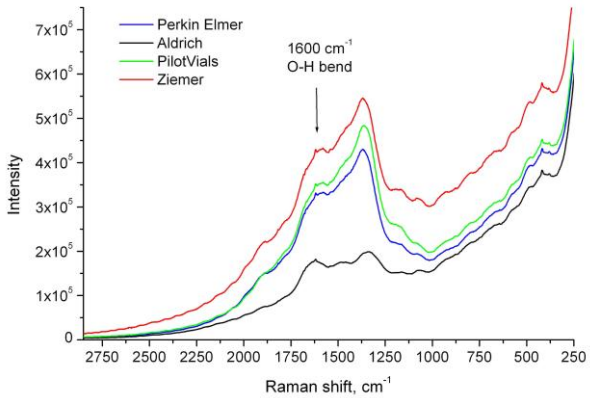


**Figure 4:** Raman spectra of water in different vials (measurement time - 15 min).

### 3.1.2. In-vial versus point-and-shoot analysis

Most of the UOC samples examined in this study yielded spectra of comparable quality when acquired in in-vial and point-and-shoot modes. There were however a few examples where the latter appeared to be a preferable option, especially when the analysis was done on bare materials. Based on the results of previous section, one may expect that the elimination of vial can bring certain benefits for samples with a weak Raman signal.

One such example is shown in Fig. 5. It relates to a uranyl hydroxide type yellow cake from Faraday mine in Canada, which exhibited a subtle molecular response over a much more intense fluorescence signal. When measured in-vial, the signal was largely dominated by the intense scattering of excitation photons on the vial's walls. On the other hand, the point-and-shoot measurement clearly showed a characteristic peak at 832 $cm^{-1}$.

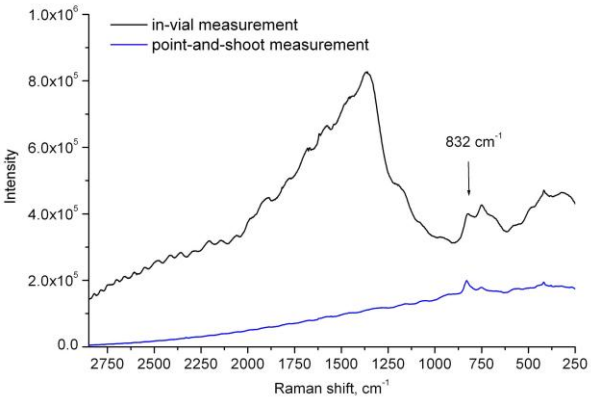


**Figure 5:** Raman spectra of a uranyl hydroxide type yellow cake from Faraday mine (Canada) acquired in the in-vial and point-and-shoot modes.

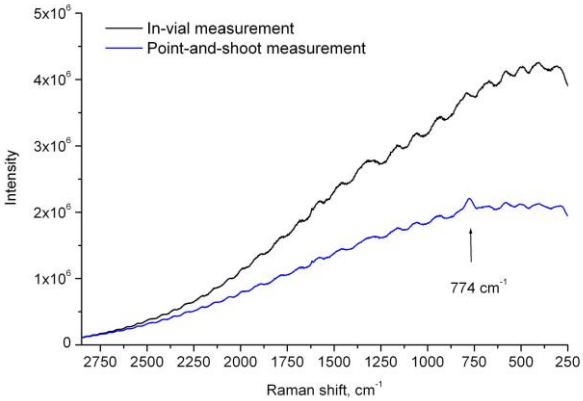


**Figure 6:** Raman spectra of an SDU type yellow cake from unspecified mine in Argentina acquired in the in-vial and point-and-shoot modes.

Another case was a yellow cake sample from an unspecified mine in Argentina. This was classified as SDU type yellow cake and based on its orange colour appeared to be readily measurable by Raman spectroscopy. The in-vial measurement, however, revealed an extremely high and fluctuating fluorescence background with no indication of molecular signal. After several attempts to measure a bare sample of this material in the point-and-shoot mode, a characteristic molecular signal at 774 $cm^{-1}$ was observed (Fig. 6). Although this latter spectrum was not a great improvement compared to the former, it did suggest that materials with a weak molecular signal could be easier detected when analysed in the point-and-shoot mode.

### 3.1.3. Pelletized powder versus loose powder

Taking into account the intended field use of the *FirstDefender RM*, any sample preparation should be minimised or ruled out. Therefore the option of pelletizing of powders was primarily explored in view of further optimization of laboratory measurements and was aimed on acquiring better quality library spectra. To this end, the powder material was pressed into pellet and then measured open in a special geometry setup that allowed rotation of the sample with respect to the direction of the laser beam. To avoid structural changes of the material, a pressure of < 2 tonne was applied.

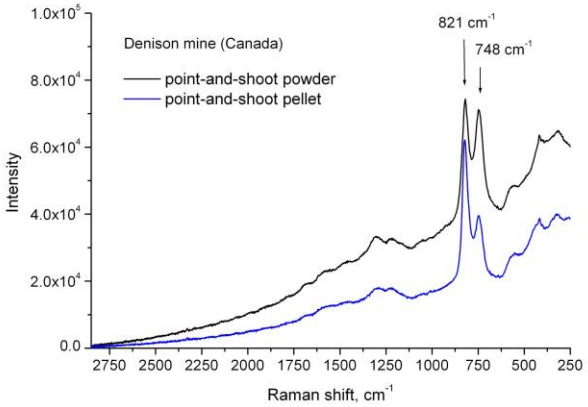


**Figure 7:** Raman spectra of an ADU yellow cake from Denison mine in Canada acquired from the powder and pressed pellet.

It was observed that the Raman signal from a pressed material is somewhat higher than that from powders. There were a few samples that delivered a sound Raman spectrum only after having been pelletized and with spectrum acquisition in the point-and-shoot mode. In a few instances the spectrum acquired from a pelletized material showed slightly different features compared to a spectrum from the same material in the powder form. One such example is shown in Fig. 7, where intensities of the closely spaced peaks show different ratios when the same ADU sample were measured as a powder and as a pellet. A possible explanation of the effect can be the better averaging of the Raman response over the sample material, which is achieved due to the rotation of the pellet during spectrum acquisition.

## 3.2. UOC samples: spectral features

### 3.2.1. Uranyl peroxide

Measurements of uranyl peroxides were rapid. Automatic scans were in the span of few seconds and molecular signals were extremely high. Library scans were also very fast with high quality spectra obtained, as the signal-to-noise ratio was in the range of 100 and above. The both in-vial and point-and-shoot measurements yielded satisfactory spectra and for the latter, the measurement of powder or pellet made no difference. A typical spectrum is shown in Fig. 8.

The characteristic Raman shifts observed included the peak at 866 $cm^{-1}$, which can be unambiguously assigned to the O-O stretching mode. The most intense peak found in the range of 828-831 $cm^{-1}$ is also unequivocal and belongs to the symmetric O=U=O stretching vibrations. The two small peaks found at the positions 355 and 280 $cm^{-1}$ can be assigned to the symmetric U-O($H_2O$) stretching and O=U=O bending modes, respectively.

### 3.2.2. Sodium diuranate

There were 5 SDU samples analysed from different locations, namely Argentina, Sweden, Niger, USA, and Romania. All the samples were orange coloured, except the sample from Romania that was dull

yellow. The latter exhibited very high fluorescent background and could not be measured.

A typical SDU spectrum is shown in Fig. 9. The main peak belonging to O=U=O stretching vibration is broad and located at lower wavenumbers (778-789 $cm^{-1}$) compared to that in peroxides. This suggests that the uranyl ion bonds are somewhat longer in this compound. The peaks in the region of 500-600 $cm^{-1}$ and at 292 $cm^{-1}$ can be assigned respectively to U-O(II) stretching and Na-O vibrations. Here II implies that O bonded to U in an equatorial plane. The peak at 292 $cm^{-1}$ may also be interpreted as belonging to a shifted O=U=O bending mode.

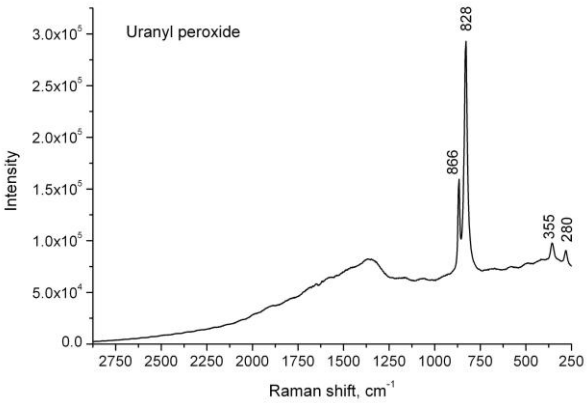


**Figure 8:** Typical Raman spectrum of the uranyl peroxide type yellow cake (Rabbit Lake, Canada).

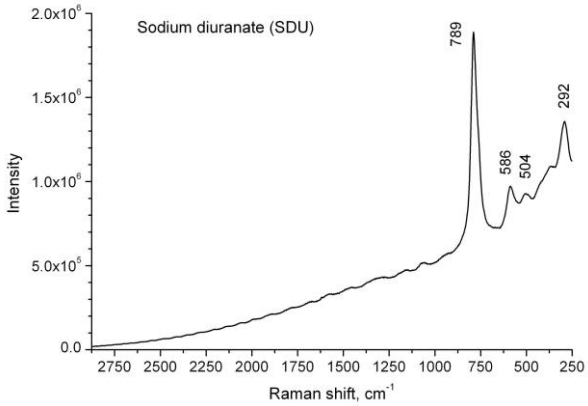


**Figure 9:** Typical Raman spectrum of the SDU type yellow cake (Somair mine, Niger).

### 3.2.3. Ammonium diuranate

This was by far one of the largest group of samples. Varying Raman responses were observed across this category of UOCs. One of the spectra is shown in Fig. 10. The main peak corresponding to the symmetric O=U=O stretch was commonly found in the region of 804-841 $cm^{-1}$. In a few cases this peak showed an internal structure of two or tree very closely spaced sub-peaks, which most likely pointed to a varying environment around the uranyl ion within the same material. The attribution of other peaks in the spectra is difficult due to the inherent complexity of the ADU.

This is also reflected in inconsistency of literature data on the composition of ammonium diuranates. Cordfunke [13] maintained that the system $UO_3$-$NH_3$-$H_2O$ embodies four discrete compounds $UO_3$-$2H_2O$, $UO_3$-$0.33NH_3$-$1.67H_2O$, $UO_3$-$0.5NH_3$-$1.5H_2O$, and $UO_3$-$0.67NH_3$-$1.33H_2O$. In contrast, Stuart and Whateley [14] indicated that ADU is described by the formula $UO_2(OH)_{2-x}$-$(ONH_4)_x$-$yH_2O$, where x varies continuously from 0 to 0.7. The above authors concluded that ammonia is bonded as $NH_4^+$, while Urbanek et al. [15] reported that part of ammonia is bonded as coordinated $NH_3$.

Despite the complexity, a thorough examination of the spectra allowed to identify signatures of several anionic impurities. Specifically, the peaks at 1097, 1047 and 1010 $cm^{-1}$ were identified as bands belonging to carbonate, nitrate and sulphate anions, respectively [16]. This was confirmed by comparing with spectra obtained with pure chemicals. Other identifiable features include the peak at 450 $cm^{-1}$, which could be interpreted as the equatorial U-O stretch, and the bands at lower wavenumbers are likely to represent bending modes of U-O.

Additionally, some of the yellow cake showed extra peaks above 1200 $cm^{-1}$. Another set of spectra, obtained for the same materials using a laboratory Raman spectrometer operated with 752.5 nm krypton laser, revealed a 550 $cm^{-1}$ shift of this group of peaks to the higher wavenumbers (Fig. 11). As the shift corresponded to the difference between the laser wavelengths, it was concluded that the peaks are related to electronic rather than Raman transitions.

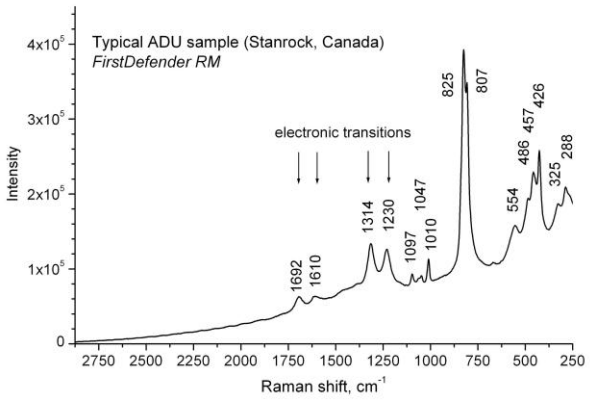


**Figure 10:** Raman spectrum of the ADU type yellow cake from Stanrock (Canada) measured with a 785 nm diode laser *FirstDefender RM*.

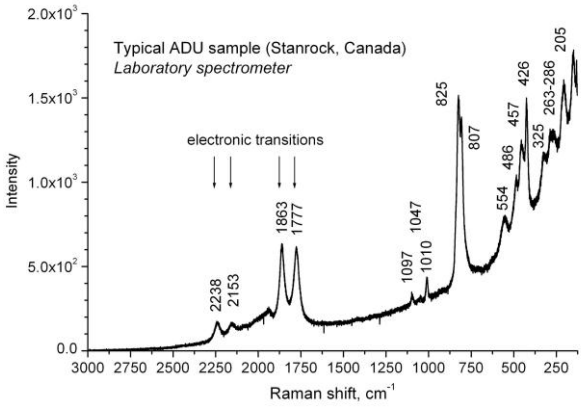


**Figure 11:** Raman spectrum of ADU sample from Stanrock, Canada, measured using a 752.5 nm krypton laser laboratory spectrometer.

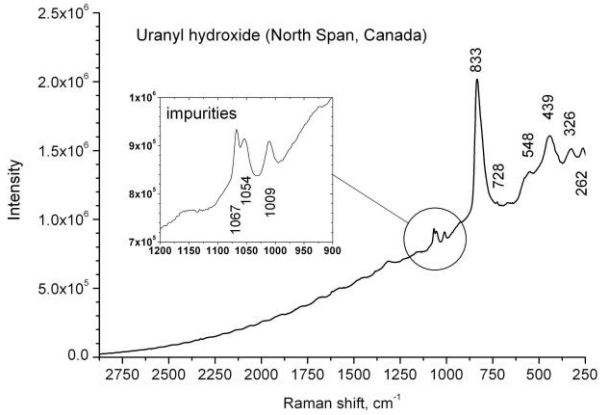


**Figure 12:** Raman spectrum of uranyl hydroxide yellow cake from North Span (Canada). The embedded graph magnifies the region with impurity signatures.

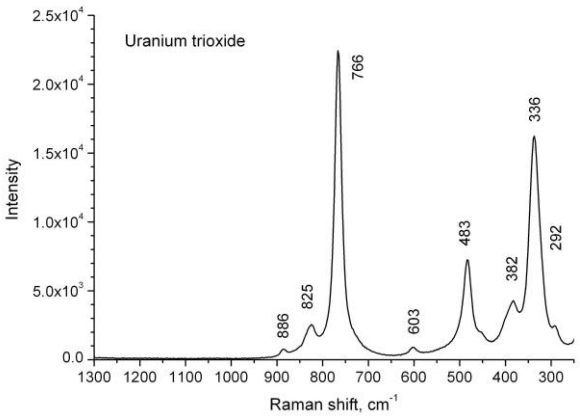


**Figure 13:** Raman spectrum of industrial uranium trioxide sample from Port Hope (Canada).

### 3.2.4. Uranyl hydroxide

The scanning times for this group of yellow cake were generally much longer due to the higher fluorescence background and weaker molecular signal (see Fig. 5). One of the best quality spectra obtained for this type of UOCs is shown in Fig. 12 (signal-to-noise ratio ~60).

In most cases, the main peak representing the symmetric stretch of uranyl ion is located in the region 810-833 $cm^{-1}$. A shoulder or second lower intensity peak was normally found in the region of 726-750 $cm^{-1}$ and can be attributed to uranyl ions being present in two non-equivalent sites. The peaks at 1067 $cm^{-1}$ and 1054 $cm^{-1}$ are related to vibrational excitations of nitrate ions, while the peak at 1009 $cm^{-1}$ is that of sulphate ions.

### 3.2.5. Uranium trioxide

There were 5 uranium trioxide samples delivered from the same facility in Port Hope, Canada. Their external appearance and similarities in their Raman spectra suggested that these were most likely different replicates of the same material. Therefore only one reference Raman spectrum was acquired for this type of yellow cake and put in the instrument's library. The spectrum is shown in Fig. 13.

It is noteworthy that the peaks in the spectrum are very sharp and intense, which is likely due to the high purity of the material. Four crystalline phases of $UO_3$ are known to exist. Literature data available for phase $\gamma$-$UO_3$ indicates the Raman peaks at 766, 690, 483, 335, 235, 103 and 51 $cm^{-1}$. These explain three strongest bands observed in Fig.13, namely at 766 $cm^{-1}$, 483 $cm^{-1}$ and 336 $cm^{-1}$.

### 3.2.6. Ammonium uranyl carbonate

As there were no industrial AUC samples available, only the synthesized ammonium uranyl carbonate was measured and included as a reference into the instrument's library. The spectrum shown in Fig. 14 is of high quality indicating a very pure material (the signal-to-noise ratio ~105). The characteristic symmetric O=U=O stretching mode appears at 816 $cm^{-1}$, again as the most intense peak in the spectrum. The peak at 1061 $cm^{-1}$ originates from stretching of the carbonate ion, whereas the peaks at 727 and 692 $cm^{-1}$ most likely represent bending modes of the same ion [17]. The ammonium does not produce any signature in the spectrum, as N-H bands are observed only above 3000 $cm^{-1}$.

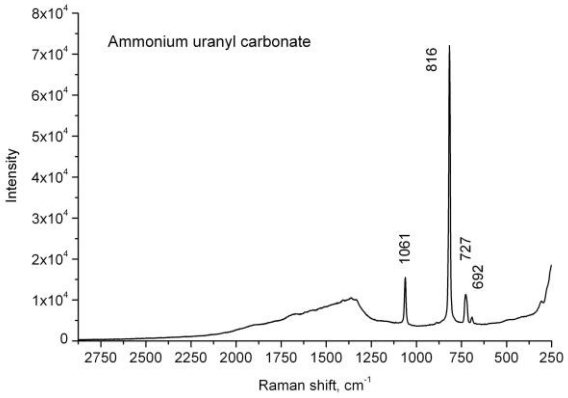


**Figure 14:** Raman spectrum of a laboratory synthesized AUC sample.

### 3.2.7. Other UOCs

The *FirstDefender RM* was incapable of measuring dark coloured samples. A quarter of the UOCs examined fell into this category and were thus not measurable. These included uranium oxides $U_3O_8$ and $UO_2$ as well as partially calcined yellowcake. From one hand, the reason for this limitation is quite clear, as dark substances largely absorb in the near infrared spectral region. On the other hand, reported recently [18] were successful measurements of $U_3O_8$ with a bench-top Raman spectrometer using the same wavelength as in the *FirstDefender RM*. Thus, further investigation is needed to identify limiting factors and potentials for extending applicability of the *FirstDefender RM* with regard to this class of compounds.

## 3.3. UOC samples: identification capability

The spectral features of different types of UOCs considered in Section 3.2 provide the basis for their identification. These features include the most intense peak associated with the symmetric stretching of uranyl ion, which is always observable in the spectrum. The position and width of this peak can be significantly affected by the presence of ligands and impurities and therefore they are characteristic of particular yellow cake's type. The ligands themselves add unique features in the spectrum and therefore they are also expected to largely contribute to the type identification. The impurity's signatures are uranium deposit and technology specific and may thus point to the origin (mine, area, plant) of yellow cake. Table 1 summarizes the spectral features observed in different types of yellow cake.

| UOC compound | Main peak | Ligand / peaks | Impurity / peaks |
|---|---|---|---|
| Uranium peroxide | 828-831 | $H_2O$ / 355 | None |
| Sodium diuranate | 778-789 | Na / 292<br>$H_2O$ / 500-600 | None |
| Ammonium diuranate | 804-841 | n/a | $CO_3^{2-}$ / 1097<br>$NO_3^-$ / 1047<br>$SO_4^{2-}$ / 1010 |
| Uranium hydroxide | 810-833 | n/a | $NO_3^-$ / 1067-1054<br>$SO_4^{2-}$ / 1008 |
| Uranium trioxide | 766 | n/a | None |
| Ammonium uranyl carbonate (synthetic) | 816 | $CO_3$ / 727, 692, 1061 | None |

**Table 1:** Characteristic spectral features of Raman responses observed in different types of yellow cake.

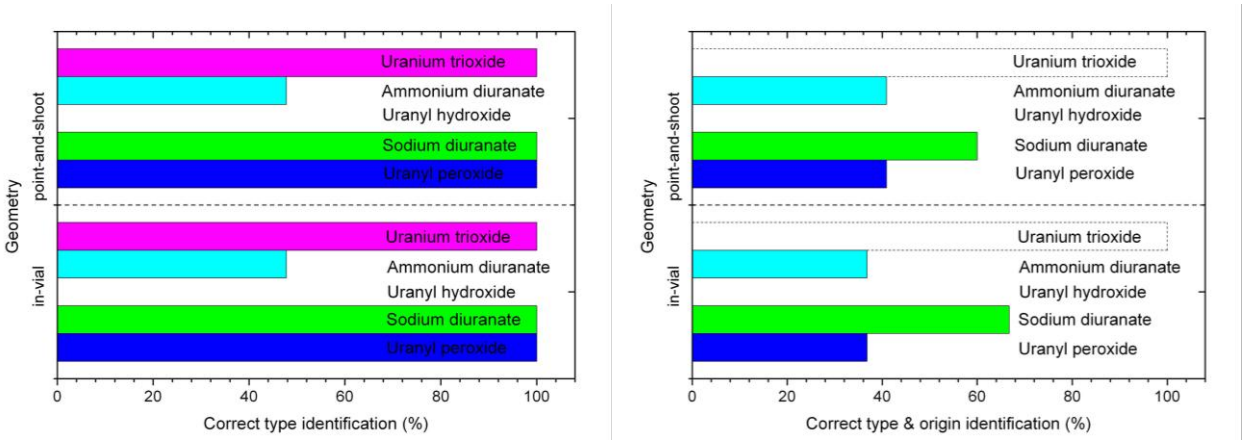


**Figure 15:** Percentage of the correct identifications of yellow cake's type (left) and yellow cake's type and origin (right). The data suggests the identification capability is rather insensitive to the measurement geometry.

As resulted from this work, the built-in library of Raman responses in the *FirstDefender RM* was extended to include some 85 UOC entries, in addition to numerous entries for variety of other materials provided in the default library. These include 46 ADU, 6 SDU, 14 uranium peroxide, 17 uranyl hydroxide, 1 uranium trioxide and 1 AUC Raman responses taken using either in-vial or point-and-shoot mode.

The identification capability of the upgraded instrument was tested on a sub-set of yellow cake samples that had been used previously to populate the library. The evaluation was performed regarding the correctness of material type identification and also regarding the correctness of the identification of both type and origin of the material. The identification of the yellow cake's type was considered correct when first material in the identification list appeared to be of correct type. The correct identification of the type and origin implied that, in addition to the previous condition, the identification list contained also the material of correct type and origin. The percentage of correct identifications in both cases is shown in Fig. 15. The relative uncertainty of the data presented is in the range of 20-30%.

The correct identification of uranyl peroxide and SDU compounds was delivered in 100% test measurements, whereas the origin of the material was correctly identified only in ~40% and ~60% cases, respectively. For both types of yellow cake the identification results could be delivered in a few seconds. No false positive identifications in total 20 trials were observed for both types of yellow cake.

The uranium trioxide samples were also identified in 100% of cases (10 correct matches out of 10 test measurements). This result however has to be considered cautiously, as the samples most likely represented replicates of the same material (see section 3.2.5). The same consideration is also valid for the identification of sample origin, as only one library entry for this type of yellow cake was available. The encouraging aspect though is that no false positive identifications were observed, thus in terms of the Raman response the material identity appears to be rather unique.

The ADU type samples could be identified in a few minutes with a decreased identification probability compared to that of the uranyl peroxide and SDU based yellow cake. Specifically, there were observed ~50% correct identifications of the material type, and about ~40% correct simultaneous identifications of the material type and origin. This means that, roughly, in 80% of cases when the ADU yellow cake is correctly identified, its origin is also accurately reported by the instrument. This is believed due to the great variability of material composition and, hence, Raman responses of different ADU samples (see Section 3.2.3). It is also noteworthy that out of 40 tests performed, only one false positive identification of the material type was reported (lead arsenide instead of ADU).

For uranium hydroxide samples the compound identification failed in all test cases, thus delivering 0% identification probability of this type of yellow cake. This is disregarding the fact that in most cases the acquired test spectrum and library spectrum were looking very similar. Moreover, there were three false positive identifications reported out of total 25 test measurements performed for this compound.

Remarkably, two of these false positives were in the case of North Span (Canada) that provided a high quality library scan shown in Fig. 12. This indicated that the actual reason for such a bad performance may lie beyond such obvious factors as high fluorescence and low molecular signal normally observed for this class of compounds.

## 4. Conclusions

The portable Raman spectrometer *FirstDefender RM* represents a mature off-the-shelf technology with promising capabilities for identification of different nuclear fuel cycle signature materials relevant to safeguards. The primary area of application of the instrument is CAs and DIVs. As resulted from the current work, the instrument's capability was extended to identify a variety of industrial yellow cake of different origin, composition and purity.

The tests showed that nearly 100% correct identification of uranyl peroxide and sodium diuranate types of yellow cake can be achieved in a few seconds. Approximately in half of the cases the origin of the material (i.e. mine or ore concentration facility) can be also correctly identified. The UOC samples containing ammonium diuranate as a major compound can be identified in a few minutes with ~50% probability. It was shown that in 80% of these cases the origin of the ADU type yellow cake can be also correctly recognized.

The other types of UOCs, including uranium dioxide, tri-uranium octa-oxide, uranyl hydroxide and some calcined yellow cake, cannot be analysed by the present version of the *FirstDefender RM*. However the literature data and recorded Raman responses suggest that at least tri-uranium octa-oxide and uranyl hydroxide yellow cake have a potential to be included in the list of identifiable compounds.

A separate study (not presented in this paper) also showed a potential of the instrument for a quick and reasonably accurate quantitative determination of the uranium concentration in uranyl nitrate solutions, which may also represent a useful capability during safeguards activities related to CAs and DIVs.

## 5. Future work

Future work will be focused on the creation of a safeguards-tailored version of the portable Raman spectrometer. This will include a further upgrade of the instrument's library with primary focus on non-radioactive safeguards-relevant proliferation indicators. A prioritized list of such material has been recently prepared by the Complementary Access Tools Working Group under the U.S. Department of Energy Next Generation Safeguards Initiative Safeguards Technology Development Program. The possibilities will be explored to further upgrade the instrument's hardware and firmware in order to extend its applicability to such "difficult" materials as uranyl hydroxides and $U_3O_8$, with the latter known to be of high safeguards relevance. An integrated capability of quantitative determination of uranium concentration in uranyl nitrate solutions will also present a useful option in the upgraded spectrometer. Exploring other advanced elemental analysis and chemical identification techniques and related off-the-shelf portable instrumentation to complement capabilities of the portable Raman spectrometer is another ongoing activity within the IAEA's Department of Safeguards, which will be pursued in the near future.

## 6. Acknowledgement

This work was done under Task A 00860 of the European Commission's Support Programme to the IAEA. The authors would like to thank Nexia Solutions for providing the UOC samples from their Springfield plant. Also the Canadian Support Programme and the Australian Support Programme to the IAEA shall be recognized for providing UOC samples.

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# Preparation of Pu particle quality control materials

**Cheolsu Kim, Takahiko Kitao, Ernesto Chinea Cano, Zsuzsanna Macsik, Stephan Vogt, Thippatai Tanpraphan, Olivier Pierre Bildstein, Laure Sangely, Jane Poths**

International Atomic Energy Agency
Department of Safeguards, Safeguards Analytical Services
IAEA Laboratories A-2444 Seibersdorf, Austria
E-mail: c.s.kim@iaea.org, kitao.takahiko@jaea.go.jp, e.chinea-cano@iaea.org, z.macsik@iaea.org, t.tanpraphan@iaea.org, o.bildstein@iaea.org, l.sangely@iaea.org, j.pohts@iaea.org

***Abstract:***

*Plutonium particles were produced at the Technical Research Centre of Finland (VTT) from a well characterized solution of CRM 136 (Department of Energy, New Brunswick Laboratory, Chicago, U.S.A.) after chemical separation of americium (Am) and the plutonium (Pu). Different particle recovery methods were tested and individual Pu particles characterized for their size, Pu content and isotopic composition. Samples with known numbers of individual particles were produced for Pu/Am age dating analysis.*

**Keywords:** particle analysis, CRM *136*, quality control particles

## 1. Introduction

The International Atomic Energy Agency's (IAEA) Environmental Sample Laboratory (ESL) carries out uranium (U) and plutonium (Pu) bulk and particle analysis of environmental samples collected during in-field safeguards activities. Environmental samples are typically swipe samples taken at nuclear facilities that are analysed to look for any indications of undeclared nuclear materialor activities carried out at these facilities. Analyses are performed by the ESL and are complemented by those of other laboratories from the IAEA's Network of Analytical Laboratories (NWAL).

Instrument calibration, method validation and quality control of the analytical processes require certified reference and well-characterized quality control (QC) materials. For U and Pu particle analytical techniques the availability of such materials is very limited.

Uranium, Pu and mixed U/Pu oxide particles were produced at the Technical Research Centre of Finland (VTT) using well-characterized solutions produced from different Certified Reference Materials (CRM) [1]. For Pu one set was produced from CRM 136 solution after chemical separation of Pu from americium (Am). Subsequently, individual $PuO_2$ particles were studied for their size distribution and other physical properties. Plutonium isotopic composition was characterized by analysing solutions produced from the dissolution of multiple particles [2].

In the particle production experiments carried out at VTT in Finland vacuum grease (Apiezon-L) was used to cover the aluminium (Al) impactor foils for better retention of the impacting particles. One technical challenge in handling the final product is the removal of particles from the Al impactor catcher foils for detailed studies and QC material production. Techniques to remove particles from the Al impactor foils and handling of individual particles for their analysis by a scanning electron microscope (SEM) and inductively-coupled plasma mass spectrometry (ICP-MS) after chemical dissolution, were tested.

The paper briefly presents the particle removal techniques, particle characterization and production of test samples for Pu/Am analysis using particle analytical techniques.

## 2. Experimental

For our particle removal tests we chose the Al impactor foil from stage 8, the same that was used in the initial characterization study by Shinonaga et al. [2]. A small section of about 1 cm was cut from the donut-shaped impactor foil and placed in a PFA (Perfluoroalkoxy, Savillex, U.S.A.) vial. After the addition of 2 mL of a solvent the vial was placed in an ultrasonic bath and sonerated for 10 minutes. Water, ethanol, isopropanol, heptane and acetone were tested in the experiment. After soneration the suspension was dried and the residue re-suspended in 0.2 mL of high purity water. A 1 µL aliquant of this suspension was applied onto the surface of a clean silicon wafer chip. The aliquant was dried and any organic residue decomposed by heating the silicon wafer to 400$^{o}$C for 20 minutes. The effectiveness of the solvents was evaluated by visual inspection of the number of particles located on the silicon wafer.

Measurement of the Pu content and isotopic composition of the individual particles was performed by ICPMS (Element II, Thermo Scientific). Individual particles were handled using a particle manipulator (Suruga Seiki, M331) attached to the microscope. Particles were transferred from the Al impactor foil directly into a pre-cleaned cryo vial and dissolved using 0.2 mL 78% $HNO_3$ (Baseline®, SEASTAR Chemicals, Canada) and 0.2 mL 45% HF (Baseline®, SEASTAR Chemicals, Canada). The solution was diluted with 2 mL high purity water prior to the ICP-MS measurements. All chemical treatments and instrumental measurements were carried out in ISO class 5 clean rooms. Direct measurements of particle size were performed using a SEM (FEI XL30, Phillips).

## 3. Results and Discussion

Bulk removal of particles from the Al impactor foil and generation of a suspension of Pu particles was studied using water, isopropanol, heptane and acetone. Acetone has proven to be most effective in removing particles from the Al impactor foil and producing a suspension with a sizeable number of distinct particles. Direct particle manipulation from Al impactor foil sector using a stainless steel needle and a micromanipulator allows easy removal and handling of discrete particles.

Individual particles from these experiments were analyzed for their size, Pu contents and isotopic composition. Table 1 summarizes the results obtained for the isotopic composition of 27 discrete $PuO_2$ particles. The isotopic composition of the generated particles agrees well with the certified values. Very small particles exhibit a larger variability in the isotopic composition attributable to the lower counting statistics for the individual Pu isotopes. No attempt was made to measure the $^{238}$Pu by alpha spectrometry and the $^{242}$Pu count rates were at or below the detection limit. The Pu concentration in these particles varies by two orders of magnitude between 0.1 and 10 pg with an average Pu content of about 2 pg. Using a density for $PuO_2$ of 11.5 g/cm$^3$ [3] we estimated the size of the particles assuming spherical geometry. Figure 2 shows the estimated size distribution of the Pu particles based on their Pu assay. It suggests that most particles removed from the impactor stage have a size between 0.4 µm and 0.8 µm. Four particles were directly measured for their size using a SEM (Figure 3). The diameter of the particles assuming spherical geometry was determined to be between 0.6 µm and 0.8 µm. Both estimates are in good agreement with the distribution obtained by Shinonaga et al. The larger number of apparently smaller particles with an estimated diameter between 0.4 µm and 0.6 µm might indicate that a larger fraction of generated particles has a density lower than the nominal value of 11.5 g/cm$^3$. Shinonaga et al. also found that the density of the two particles analysed is considerably lower (5 g/cm$^3$ – 9 g/cm$^3$) than the nominal density of $PuO_2$.

A select number of particles were manipulated and transferred onto a graphite sample holder, either as distinct particles or particle doubles, for further studies of their Pu and Am isotopic contents and their nominal $^{241}$Pu-$^{241}$Am age.

| No. | $^{240}Pu/^{239}Pu$ [at/at] | σ | $^{241}Pu/^{239}Pu$ [at/at] | σ | Total Pu (pg) |
|---|---|---|---|---|---|
| 1 | 0.145 | 0.002 | 0.008 | 0.000 | 5.42 |
| 2 | 0.146 | 0.001 | 0.008 | 0.000 | 10.32 |
| 3 | 0.145 | 0.003 | 0.008 | 0.001 | 2.39 |
| 4 | 0.141 | 0.003 | 0.007 | 0.001 | 2.94 |
| 5 | 0.142 | 0.003 | 0.007 | 0.001 | 2.52 |
| 6 | 0.146 | 0.003 | 0.008 | 0.001 | 2.96 |
| 7 | 0.144 | 0.003 | 0.007 | 0.001 | 2.30 |
| 8 | 0.149 | 0.003 | 0.008 | 0.001 | 2.06 |
| 9 | 0.140 | 0.003 | 0.007 | 0.001 | 2.27 |
| 10 | 0.145 | 0.003 | 0.008 | 0.001 | 2.30 |
| 11 | 0.145 | 0.005 | 0.008 | 0.001 | 0.98 |
| 12 | 0.135 | 0.014 | 0.005 | 0.004 | 0.11 |
| 13 | 0.144 | 0.005 | 0.008 | 0.001 | 0.86 |
| 14 | 0.147 | 0.014 | 0.009 | 0.003 | 0.11 |
| 15 | 0.145 | 0.014 | 0.008 | 0.003 | 0.12 |
| 16 | 0.151 | 0.014 | 0.011 | 0.004 | 0.11 |
| 17 | 0.145 | 0.014 | 0.010 | 0.005 | 0.11 |
| 18 | 0.146 | 0.005 | 0.007 | 0.001 | 0.98 |
| 19 | 0.143 | 0.005 | 0.008 | 0.001 | 0.99 |
| 20 | 0.143 | 0.005 | 0.008 | 0.001 | 1.05 |
| 21 | 0.142 | 0.005 | 0.007 | 0.001 | 1.01 |
| 22 | 0.143 | 0.007 | 0.007 | 0.002 | 0.48 |
| 23 | 0.143 | 0.010 | 0.008 | 0.002 | 0.89 |
| 24 | 0.152 | 0.008 | 0.008 | 0.002 | 1.26 |
| 25 | 0.147 | 0.011 | 0.008 | 0.002 | 0.70 |
| 26 | 0.141 | 0.007 | 0.008 | 0.002 | 1.63 |
| 27 | 0.148 | 0.011 | 0.009 | 0.002 | 0.79 |
| Cerified. Value | 0.145 | 0.0002 | 0.0067 | 0.00002 | |

**Table 1:** Pu isotopic ratios and total Pu content in single particles measured by ICP-MS.

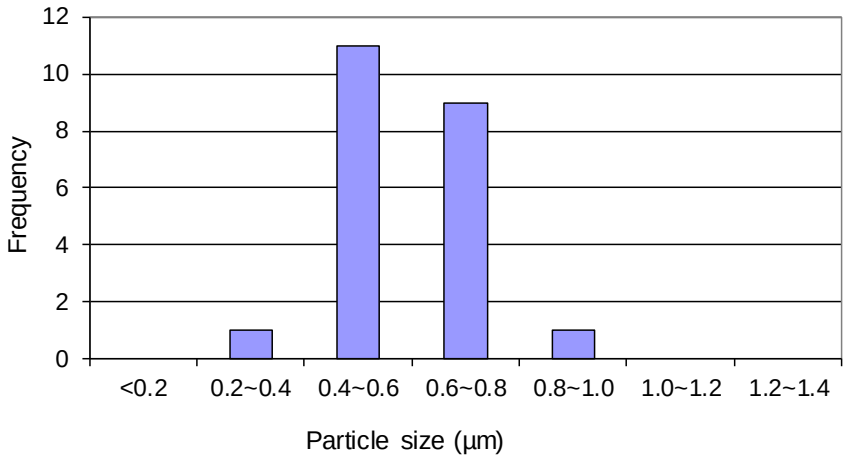


**Figure 2:** Size distribution of Pu particles.

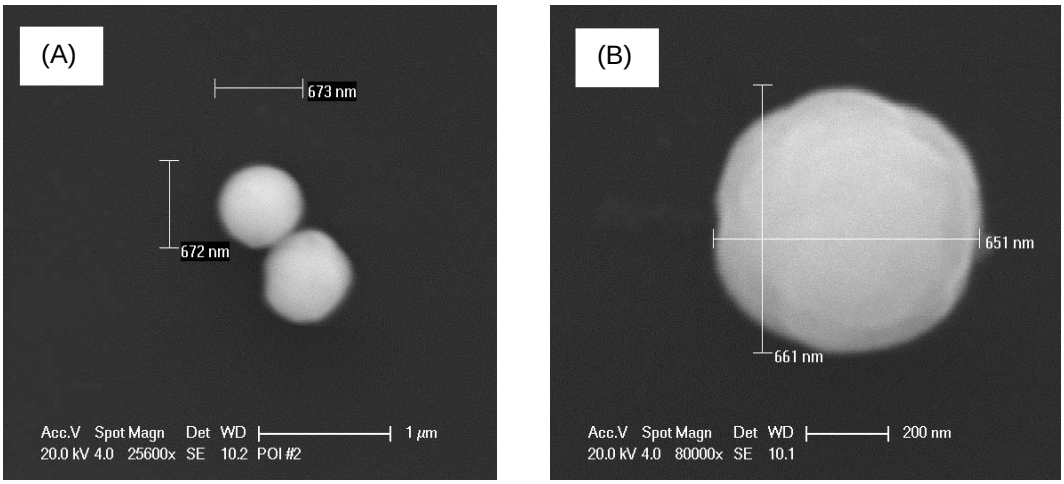


**Figure 3:** SEM images of a double Pu particles (A) and a single Pu particle (B).

## 4. Conclusions

Micrometer-sized Pu particles were produced in an aerosol generator (TSI 3076, TSI Inc.) and collected on Al impactor foils treated with Apiezon-L for efficient particle retention. An effective technique for particle removal from the Al impactor stage uses ultrasoneration in acetone and drying on an appropriate substrate such as a silicon wafer for further handling.
Individual particles from this experiment were handled using a micromanipulator in combination with a digital microscope and measured for their size, Pu content and isotopic composition.
The average size of particles determined from direct SEM measurements is 0.6 – 0.8 µm. The size of Pu particles calculated from their Pu assay (0.1 pg to 10 pg) and assuming spherical geometry and a $PuO_2$ density of 11.5 g/cm$^3$ ranges between 0.4 µm and 0.8 µm. This suggests that a larger fraction of the generated particles did not reach the nominal density of $PuO_2$ in agreement with the initial characterization. The isotopic ratios of $^{240}Pu/^{239}Pu$ and $^{241}Pu/^{239}Pu$ are in good agreement with previous measurements and the certified values from the certificate of CRM 136.

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# Simple and fast Pu separation for 'fast' Pu screening purposes

**Zsuzsanna Macsik, Cheolsu Kim, Stephan Vogt**

International Atomic Energy Agency
Department of Safeguards, Safeguards Analytical Services
IAEA Laboratories A-2444 Seibersdorf, Austria
E-mail: z.macsik@iaea.org, c.s.kim@iaea.org, k.vogt@iaea.org

***Abstract:***

*Bulk analysis of safeguards swipe samples performed at the Environmental Sample Laboratory (ESL) of the International Atomic Energy Agency's (IAEA) Safeguards Analytical Laboratories provides uranium (U) and plutonium (Pu) content and isotopic information. Initially, after digestion, the samples are screened to estimate the U and Pu content and to determine the amount of U and Pu tracers and sample aliquot to be used in the analysis. The U content can be estimated directly by inductively coupled plasma mass spectrometry (ICP-MS) from diluted sample solution, while chemical separation is required for Pu quantification. Several separation methods have been tested in order to simplify and shorten the Pu screening procedure while achieving the desired performance parameters, e.g. recovery, U decontamination factor and interferences.*

**Keywords:** bulk analysis; swipe samples; Pu; ICP-MS

## 1. Introduction

Environmental sampling was first introduced as a safeguards strengthening measure in 1996. It is now in routine use as a safeguards verification measure and is a powerful tool for detecting undeclared nuclear material and activities at declared facilities and other locations. Swipe sampling is the most efficient means for inspectors to collect environmental samples during in-field verification activities. The samples are analysed at the Environmental Sample Laboratory (ESL) of the International Atomic Energy Agency's (IAEA) Safeguards Analytical Laboratories and other qualified laboratories in the Network of Analytical Laboratories (NWAL) in order to determine the following information: (1) uranium (U) and plutonium (Pu) concentration and isotopic composition, (2) the ratio of americium (Am) and Pu isotopes, and (3) U and Pu isotopic composition of single particles in case of particle analysis [1,2].

Environmental samples are initially screened by gamma spectrometry to detect fission and decay products and by X-ray fluorescence spectrometry to detect and locate the presence of U and Pu. A suitable analytic method is chosen according to the screening results: the samples can be examined by either bulk (inductively coupled plasma mass spectrometry (ICP-MS), thermal ionization mass spectrometry (TIMS)) [3], particle analysis (scanning electron microscopy (SEM), secondary ion mass spectrometry (SIMS)) [2], or a combination of these.

For U and Pu bulk analysis, a radioanalytical method based on ion exchange and extraction chromatography is currently used to separate and purify U and Pu prior to mass spectrometric measurements. After sample digestion (ashing and acid digestion), sample splitting and spiking, the U and Pu are separated using a two-column-system of AG MP-1® ion exchange and UTEVA® extraction chromatographic columns. After separation, each fraction is purified further by ion exchange using AG MP-1®. The separated and purified fractions are measured by TIMS and ICP-MS. The whole procedure is usually performed in 14 to 18 working days.

After digestion, the samples are screened to estimate the U and Pu amounts in the samples to determine the amount of U and Pu tracers and sample aliquot to be used in the analysis. For U screening 0.05%-0.1% of the sample solution is used for measurement by ICP-MS. In the case of Pu screening, Pu needs to be separated mainly from U prior to the ICP-MS measurement.

After spiking, the sample aliquots are evaporated to dryness and the residue is dissolved in 4 mL 8 mol $L^{-1}$ $HNO_3$ and loaded on an AG MP-1® column. After rinsing, the column with 50 mL 8 mol $L^{-1}$ $HNO_3$ and 20 mL 11 mol $L^{-1}$ HCl Pu is eluted with 20 mL 47 w/w% HBr. The Pu fraction is evaporated to dryness which takes approximately 10 hours. In total, the whole Pu screening procedure takes two working days with overnight evaporation.

The main objective of this study was to shorten this procedure while still meeting the required performance parameters, such as Pu recovery and U decontamination factor.

The aim of Pu screening is to get preliminary information on the presence of Pu and its content in the samples. Typical Pu content in swipe samples are in the picogram to femtogram range and the applied separation procedure has to guarantee high Pu recovery and U decontamination factor. Besides the performance parameters, it is also a strong requirement to use the least amount of reagents. Obtaining results with high precision is not a crucial requirement since Pu screening gives only an estimation on the Pu content.

Separation of Pu from other actinides and matrix elements is very well studied in the literature. Ion exchange and extraction chromatography are mainly used in cases of environmental samples. In this study AG MP-1® anion exchange resin and TEVA extraction chromatographic resin were investigated.

Several oxidation state adjustment procedures and Pu eluents have been tested, compared and applied to ion exchange and extraction chromatographic methods. The current Pu screening procedure has been investigated, improved and compared with several extraction chromatographic methods.

## 2. Experimental

### 2.1. Equipment

All measurements were carried out by a magnetic sector field ICP-MS using an Element 2 ICP-MS (Thermo Electron Corp., Germany) in low resolution mode with a high efficiency inlet system, Apex HF (Elemental Scientific, USA) in self-aspirating mode.

### 2.2. Reagents, materials and samples

The anion exchange resin used was AG MP-1® (BioRad® 50-100 mesh, active group: $R\text{-}CH_2N^+\text{-}(CH_3)_3$). The extraction chromatographic resin used was TEVA (TRISKEM International SAS, 100-150 µm particle size, alphatic quaternary amine: trialkyl, methylammonium nitrate (or chloride)). The resins were always placed in plastic BioRad columns.

For tracer solutions, $^{242}Pu$ (Spike Isotopic Reference Material IRMM 085, Geel, Belgium) and $^{233}U$ (NIST Standard Reference Material (SRM) 995, Gaithersburg, USA) were used.

All reagents used (Mohr's salt, hydrazine, $NaNO_2$, $Fe(NO_3)_3x9H_2O$, ascorbic acid) were of analytical grade. Acids used ($HNO_3$, HCl, HF) and $H_2O_2$ were high purity chemicals (Baseline® grade) (Seastar Chemicals INC, Canada). Water used was ultrapure water (Milli-Q System, Millipore, USA).

Model samples were prepared from tracer solutions of $^{242}Pu$ and/or $^{233}U$ in $HNO_3$ or HCl. The quality cvontrol (QC) swipe samples were prepared in 1999 using either CRM U030-A U isotopic standard or CRM 136 Pu isotopic standard (New Brunswick Laboratory, Argonne, USA). Because the samples

analysed were actual swipe samples collected by inspectors during safeguards inspections, no isotopic information is published.

## 2.3. Separation methods tested

### 2.3.1. Tests of ion exchange methods – Improvement of the original procedure

Firstly, the original procedure, as described in the Introduction, was investigated and Pu recovery and U decontamination factor were determined. Every test was performed using model samples.

Secondly, several Pu eluents which could be measured directly by ICP-MS were tested focusing on the Pu recovery: 0.5 mol $L^{-1}$ HCl, 0.5 mol $L^{-1}$ HCl / 0.1 mol $L^{-1}$ HF and 0.1 mol $L^{-1}$ HCl / 0.1 mol $L^{-1}$ HF. In case of short, effective elution of the Pu with these eluents (in 2-4 mL), the ICP-MS measurement could have been done directly eliminating the evaporation of larger volumes of acids.

Thirdly, the original eluent (47% HBr) was investigated. Elution of Pu with HBr is based on two processes: formation of bromide complexes and reduction of Pu(IV) to Pu(III). We assumed that the kinetics of these processes for complete Pu elution require more time than is available while 10 mL HBr runs through the column twice (approximately 5 min). Therefore, to improve the Pu elution with HBr (increase the Pu recovery and decrease the volume of the eluent), only a small quantity of HBr (200 or 500 µL) was added at a time onto the column and different waiting times were applied, i.e. (1) 4-5 min after reaching each 4 mL eluent, (2) 1 min after reaching 4 mL eluent, (3) approximately 1 min after each 500 µL portion and 4-5 min after reaching 2 mL eluent. Plutonium recovery, Pu elution curves and U decontamination factors were determined.

Finally, the modified Pu screening procedure was tested using QC and real swipe samples.

### 2.3.2. Tests of extraction chromatographic methods

In the case of extraction chromatographic methods, various oxidation state adjustment procedures and different Pu eluents were performed under the same separation scheme. Only the $HNO_3$ (or HCl) concentration of the loading solution varied depending on the required concentration of the selected oxidation state adjustment procedure. The BioRad columns were always filled with 1.4 mL TEVA resin (previously soaked in $H_2O$). The column was washed with 10 mL $H_2O$, 10 mL 9 mol $L^{-1}$ HCl and conditioned with 20 mL $HNO_3$ (or HCl) with the loading solution's concentration. After loading the sample solutions, the columns were washed with 20 mL 4 mol $L^{-1}$ $HNO_3$, followed by 10 mL 9 mol $L^{-1}$ HCl. Subsequently the Pu was eluted from the column using various eluents.

The following oxidation state adjustment procedures were applied and compared:

- The $HNO_3$ concentration of the samples were adjusted to 8 mol $L^{-1}$. This procedure was tested by applying various waiting times prior to loading and brief heating to different temperatures [10,11]. (Called the "8 mol $L^{-1}$ $HNO_3$" procedure.)
- The $HNO_3$ concentration of the samples was adjusted to >6 mol $L^{-1}$, 100 mg of $NaNO_2$ was added and the samples were heated mildly [21]. (Called the "$NaNO_2$" procedure.)
- Approximately 50 mg of Mohr's salt was added to the samples in 1 mol $L^{-1}$ $HNO_3$. Iron and all Pu species were reduced with $N_2H_4xH_2O$ to Fe(II) and Pu(III), respectively (the presence of Fe(III) was checked by the rodanide test). After complete reduction, the $HNO_3$ concentration was adjusted to 4 mol $L^{-1}$ and the samples were boiled to destroy the hydrazine excess and Fe was oxidized to Fe(III) and Pu(III) to Pu(IV). Finally Pu(IV) was stabilized by adding approximately 60 mg of $NaNO_2$ to the samples. ([15, 16] adapted to TEVA resin). (Called the "Fe(II)-$N_2H_4xH_2O$-$NaNO_2$" procedure.)

- Different amounts (3-28 mg) of Fe(III) in nitrate form and ascorbic acid (50-100 mg) were added to the samples and the $HNO_3$ concentration of the samples was adjusted to 5 mol $L^{-1}$ [13]. (Called the "Fe(III)-ascorbic acid" procedure.)
- Approximately 13 mg Fe(III) in nitrate form was added to the samples. Iron and all Pu species were reduced with $N_2H_4xH_2O$ to Fe(II) and Pu(III), respectively (the presence of Fe(III) was checked by rodanide test). After complete reduction the $HNO_3$ concentration was adjusted to 4 mol $L^{-1}$ and the samples were boiled to destroy the hydrazine excess. Approximately 200 mg of $NaNO_2$ was added to the samples and boiled again to destroy the $HNO_2$ excess in order to prevent bubble formation on the column. After cooling the samples, the $HNO_3$ concentration was adjusted to 8 mol $L^{-1}$. [14] (Called the "Fe(III)-$N_2H_4xH_2O$-$NaNO_2$" procedure.)
- Hydrogen peroxide was also tested as a redox reagent in a HCl medium. A few drops of $H_2O_2$ were added to 11 mol $L^{-1}$ HCl medium and the samples were rested for approximately 45 min prior to loading [21]. (Called the "HCl-$H_2O_2$"procedure.)

In addition to the oxidation state adjustment procedures, different Pu eluents were also tested: 0.1 mol $L^{-1}$ HCl, 0.5 mol $L^{-1}$ HCl, 0.1 mol $L^{-1}$ HCl / 0.1 mol $L^{-1}$ HF and HBr for the same purpose as in chapter 2.3.1.

### 2.4. The improved Pu screening procedure

Following the standard procedure for determination of U and Pu quantities in safeguards swipe samples after ashing and acid digestion, the sample residues were dissolved in 8 mL 8 mol $L^{-1}$ $HNO_3$. Approximately 5% of the samples and approximately 700 pg $^{242}Pu$ tracer were used for Pu screening. The spiked sample aliquots were evaporated to dryness, dissolved in 4 mLmL 8 mol $L^{-1}$ $HNO_3$ and heated at 100-110 ºC (just below the boiling point) for a short time. After at least 60 min they were loaded into a column containing 1.4 mL AG MP-1® ion exchange resin pre-conditioned by 30 mL 8 mol $L^{-1}$ $HNO_3$. The beakers containing the load solutions were rinsed with 2 mL 8 mol $L^{-1}$ $HNO_3$ three times and then the column was rinsed with 50 mL 8 mol $L^{-1}$ $HNO_3$ and 20 mL 11 mol $L^{-1}$ HCl. Plutonium was eluted slowly with 4 mL 47% HBr by adding only 500 µL portion at a time and waiting 4-5 min after reaching the first 2 mL. The Pu fraction was then evaporated to dryness. It was dissolved in 300 µL 15 mol $L^{-1}$ $HNO_3$ along with a few drops of $H_2O_2$ and evaporated to dryness again. This procedure was repeated and the residue finally dissolved in 2 mL 2% $HNO_3$ for ICP-MS measurement.

The improved Pu screening procedure can be performed in one working day.

## 3. Results and discussions

### 3.1. Tests of ion exchange methods

The original Pu screening procedure typically achieved Pu recovery of 60%-70% and U decontamination factors of 2500-4000.

Plutonium eluents directly measurable by ICP-MS, such as 0.5 mol $L^{-1}$ HCl, 0.5 mol $L^{-1}$ HCl / 0.1 mol $L^{-1}$ HF and 0.1 mol $L^{-1}$ HCl / 0.1 mol $L^{-1}$ HF, resulted in poor Pu recoveries (varying between 28% and 47%) and U decontamination factors (1000-1300) even using larger volumes of the eluent (10-15 mL). Low Pu recovery and using larger amount of directly measurable eluents resulted in significantly higher detection limits for Pu. Therefore these eluents cannot be applied for Pu screening purposes.

Slow elution of Pu with 47% HBr gave acceptable Pu recoveries and U decontamination factors. Adding the eluent in 200 µL portions and waiting 4-5 min after reaching 4 mL eluent allowed nearly 100% Pu recovery using 20 mL 47% HBr. It was observed that the Pu elution profile depended to a high degree upon which elution technique was applied. Figure 1 shows two examples of Pu elution profiles obtained by varying the elution rate.

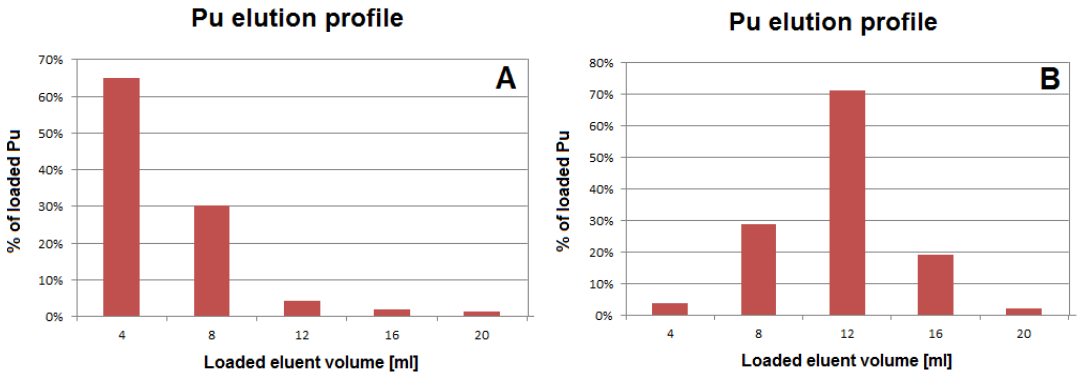


**Figure 1:** The obtained Pu elution profiles using different slow elution techniques (A: adding the eluent in 200 µL portions and waiting 4-5 min after reaching 4 mL eluent; B: adding the eluent in 500 µL portions and waiting approximately 1 min after reaching 4 mL eluent)

Evidently, it is very important to elute the Pu slowly to minimize the eluent volume. Good Pu recoveries (85%-102%) and U decontamination factors (5400-6300) could be achieved by eluting the Pu with 4 mL 47% HBr by adding 500 µL portions at a time and waiting 4-5 min after reaching 2 mL eluent.

## 3.2. Tests of extraction chromatographic methods

The so called "8 mol $L^{-1}$ $HNO_3$" oxidation state adjustment procedure was optimized. The adjustment of the $HNO_3$ concentration to 8 mol $L^{-1}$ without heating resulted in significantly lower Pu recoveries compared to the procedures where short, relatively high temperature heating was applied. According to the optimized procedure, the samples are heated at 100-110 ºC for a short time (10-15 min) and rested for at least 60 min prior to loading.

The Pu recoveries obtained by the oxidation state adjustment procedures listed in chapter 2.3.2. were comparable or significantly lower than those of obtained by using the "8 mol $L^{-1}$ $HNO_3$" procedure. The U decontamination factors by the oxidation state adjustment procedures listed in chapter 2.3.2. were comparable or slightly better than those obtained by using the "8 mol $L^{-1}$ $HNO_3$" procedure with one exception. Using the "Fe(III)-$N_2H_4xH_2O$-$NaNO_2$" procedure, the achieved Pu recovery was comparable but the U decontamination factor was one order of magnitude higher than that obtained by the "8 mol $L^{-1}$ $HNO_3$" procedure. Using 10 mL 47% HBr as Pu eluent, the Pu recoveries were 60%-70% and 55% and the U decontamination factors were $10^4$ and $10^5$ obtained by the "8 mol $L^{-1}$ $HNO_3$" procedure and the "Fe(III)-$N_2H_4xH_2O$-$NaNO_2$" procedure, respectively.

The directly measurable Pu eluents, i.e. 0.1 mol $L^{-1}$ HCl, 0.5 mol $L^{-1}$ HCl and 0.1 mol $L^{-1}$ HCl / 0.1 mol $L^{-1}$ HF, gave poor Pu recoveries (< 40%) and U decontamination factors (< 1300). These results were similar to those of the tests of ion exchange methods (see chapter 3.1.).

The elution with *diluted HCl* is based on the fact that the capacity factor (k') of Pu in diluted HCl (< 2 mol $L^{-1}$) is lower than 10, so Pu can be theoretically eluted from the column. Unfortunately U retains on the TEVA column in more concentrated HCl (> 5 molL$L^{-1}$), as well and the k' of U is low (< 1) in diluted HCl (< 0.5 mol $L^{-1}$). This can be the reason why the U decontamination factors were so low.

Additional experiments showed that while using only diluted HCl as Pu eluent, Pu cannot be completely removed from the TEVA column: The effluent (the loading solution passed through the TEVA column) and the rinsing solutions (20 mL 4 mol $L^{-1}$ $HNO_3$ and 10 mL 9 mol $L^{-1}$ HCl) contained approximately 8% of the loaded Pu. In the Pu fraction (10 mL 0.1 mol $L^{-1}$ HCl) only 60% of the loaded Pu could be found. It was assumed that the missing Pu (32%) could not be eluted from the TEVA column.

The elution with *0.1 mol $L^{-1}$ HCl / 0.1 mol $L^{-1}$ HF* is based on the formation of fluoride complexes with Pu(IV). This complex formation is unfortunately not selective since U(VI) also have fluoride complexes. Therefore, acceptable U decontamination factors cannot be achieved using this eluent either.

As mentioned before, the elution with *HBr* is based on not only the formation of bromide complexes, but the reduction of Pu(IV) to Pu(III), as well. The latter process is selective since U would retain on the TEVA column even in case of reduction of hexavalent U to tetravalent. Therefore, a high U decontamination factor can be achieved. In the case of complete Pu reduction, high Pu recovery can also be assumed. Unfortunately the kinetics of this process using HBr is relatively slow, according to our experimental results. The optimal solution would be to use reducing agents (e.g. $NH_4I$ [9,21], KI [24], HI [10], $TiCl_3$ [12,15,16], hydroquinone [20], ascorbic acid [12] or a mixture of ascorbic acid and $Fe(NH_2SO_3)_2$ [12]) in 9 mol $L^{-1}$ HCl to elute Pu selectively as Pu(III) while U stays on the TEVA column. This can guarantee high Pu recoveries and U decontamination factors due to the selectivity of the reduction process. However it would sacrifice our effort to shorten the Pu screening procedure since reagents from the Pu fraction should be removed by additional chemical treatment prior to ICP-MS measurement.

### 3.3. Analysis of QC and real swipe samples

The U and Pu QC and inspection swipe samples were prepared according to the Pu screening procedure described in chapter 2.4. The U QC swipe sample was measured to determine the achievable Pu recoveries and U decontamination factors by analysing real samples. The other samples were analysed to test the capabilities of the procedure.

Analysing two sets of U QC swipe samples showed that the modified Pu screening procedure meets all the requirements described in chapter 1: (1) high Pu recoveries (85% and 102%), (2) acceptable U decontamination factors (5400 and 6300) , (3) only small volumes of very few reagents ($HNO_3$, HCl, HBr and $H_2O_2$, no redox reagent) and (4) completion of the analysis within one working day.

Table 1 shows the Pu amount detected in the swipe samples.

| | **Pu [pg]** | **Pu recovery [%]** |
|---|---|---|
| **Pu QC swipe** | 0.142 ± 0.018 | 88% |
| **Inspection-A** | 0.028 ± 0.011 | 85% |
| **Inspection-B** | 0.027 ± 0.012 | 93% |
| **Inspection-C** | 0.009 ± 0.008 | 94% |

**Table 1:** Results of the analysis of Pu QC and inspection swipe samples

As the interference effect of W, Os, Hg and Pb was less than 0.001 cps (the contribution of the interference elements to the $^{239}Pu$ and $^{240}Pu$ intensities), the interference correction was not performed. However, U interference correction (including abundance sensitivity at m/z = 239 and m/z = 240 and $UH^+$ formation at m/z = 239) was applied.

The results of the inspection swipe samples could not be obtained with a high degree of accuracy due to the low Pu amount in the analysed sample aliquots. It should be noted that the results refer to the whole swipe sample. The detected Pu amounts in the measured sample aliquots (~5%) were at the low fg level, close to the Pu detection limit.

The results of the Pu QC swipe compared well with the results of a previous measurement of this type of QC sample (Pu amount: 0.139 ± 0.008) within the uncertainties.

## 4. Conclusions

An improved Pu screening method was established and validated, meeting all the defined requirements of Pu recovery, U decontamination factor, reagents used and analysis time. Although similar Pu recoveries and 2-3 times higher U decontamination factors could be achieved by the separation method based on TEVA extraction chromatography, this method would not shorten the Pu screening which was our main objective. In cases of extremely high U/Pu ratios (>$10^9$) its use can still be considered to obtain more precise estimation on the Pu content of the swipe samples.

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# Production of monodisperse uranium particles for nuclear safeguards applications

**Alexander Knott, Martin Dürr**

Forschungszentrum Jülich
IEK-6: Nuclear Waste Management and Reactor Safety
52425 Jülich (Germany)
E-mail: a.knott@fz-juelich.de, ma.duerr@fz-juelich.de

***Abstract:***

*Particle analysis of nuclear materials is a powerful tool in nuclear safeguards and nuclear forensic studies. For the analysis quality control and reference material samples are required for instrument and method validation as well as the assurance of the quality of reported results. The availability of uranium- and plutonium-containing particles with well-defined properties such as size, density, elemental and/or isotopic composition is limited.*

*The potential of particle production processes starting with monodisperse aerosols prepared from dilute uranium/plutonium solution has already been demonstrated. Monodisperse uranium and plutonium particles with known elemental contents can be produced using a Vibrating Orifice Aerosol Generator (VOAG). The production of these particles is a two-step process: 1) generation of uniform aerosol particles, and 2) drying, calcining and collection of particles.*

*This paper describes an experimental set-up for particle generation and issues associated with the overall particle production process, its current status and future plans for production of the various nuclear materials needed.*

**Keywords:** Environmental Sampling; Destructive Assay; Isotopic Measurements; Nuclear Forensic; LG-SIMS

## 1. Introduction

Environmental sampling is administered in states as part of the Safeguards Agreements with the International Atomic Energy Agency (IAEA) and provides a tool to strengthen the effectiveness and improve the efficiency of the safeguards system. It is a powerful measure to investigate the completeness of a State's compliance with its safeguards agreements; Environmental sampling is also part of IAEA's legal authority stemming from the Model Additional Protocol (INFCIRC/540). Generally, environmental swipe sampling has been implemented by the IAEA by collecting swipe samples from within nuclear facilities. The swipe samples are investigated specifically for the presence of undeclared nuclear materials and activities. It is fundamentally based upon the fact that every industrial process releases traces such as particulates into the environment. These signatures are representative for a facility's activities and can be found even decades after a specific process has been discontinued.

Swipe samples returned from inspections are initially screened for traces of nuclear activities such as fission and activation products using radiometric methods. Analyses using bulk techniques provide more detailed information on the presence, concentration and isotopic composition of U, Pu and other safeguards relevant information. They provide, however, only average results for the entire sample.

Complementary to the bulk analytical efforts, particle analysis has emerged as an important analytical tool for nuclear safeguards verification activities as well as forensic investigations over the past decade. Particle analysis refers to the characterization of discrete micro- and sub-micrometer-sized particles and properties such as their micro-crystallography, morphology, elemental and isotopic

composition are determined. For the measurement of these properties a variety of instrumental techniques are employed such as Scanning Electron Microscopy (SEM) with or without energy dispersive and/or wavelength dispersive X-ray Fluorescence Spectrometers (EDX/WDX), or Thermal Ionization and Secondary Ion Mass Spectrometry (TIMS and SIMS).

SIMS is used in safeguards and forensic investigations to identify, for example, uranium containing particles at the same time analyzing them for their isotopic composition. Evaluation and calibration of SIMS and other instrumental methods requires reference and Quality Control (QC) materials. Ideally, such particles resemble those analyzed from inspection samples. However, one needs to concede to the fact, that a wide variety of particles exist on collected swipe samples and that reference material (RM) can be synthesized only in limited types and characteristics. For reference particles it would be desirable to be able to tailor the particle properties according to the need of analytical procedures to be applied in particle analysis.

Different approaches to produce QC particles have been used in the past. At the VTT Technical Research Centre in Finland particles were produced from certified reference materials using an atomizer as particle generator. [1] These particles have a well characterized isotopic composition, but cover a wide range of sizes and densities. At the Institute for Reference Materials and Measurement (IRMM) in Belgium, particles were produced by hydrolyzing uranium hexafluoride in a controlled environment [2]. Uranium oxyfluoride and oxide particles of various sizes and compositions were produced in these experiments, the particles closely matching material found in enrichment facilities or facilities handling uranium hexafluoride. Special glass micrometer sized particles with varying amounts of uranium and uranium isotopics were produced at the IRMM [3], as were uranium oxide particles produced at Harwell for QC purposes [4]. Monodisperse particles were produced and investigated at the Joint Research Centre's Institute for Transuranium Elements in Karlsruhe, Germany. Using a Vibrating Orifice Aerosol Generator (VOAG) a study was carried out to produce particles of well-defined size, density, geometry and elemental contents [5] [6] [7].

This paper describes a setup under development for the production of monodisperse particles using a VOAG with the goal to establish a robust and stable production process. This paper presents the current status and results obtained by the first test-runs.

## 2. Particles for Quality Control and Quality Assurance

For particles to be suitable for quality control and quality assurance, certain properties need to be fulfilled. As with any reference material, homogeneity within the sample needs to be ensured and verified. For particles this means, that particles should have uniform properties within the batch of produced particles to the furthest degree which is possible. With a batch of uniform particles, identical test particles can be provided to the various users of analytical methods. In principle, the property of every individual particle needs to be known, which is verified by characterizing single particles based on a statistical sampling scheme. A number of techniques and procedures to identify and analyze individual particles exist. If morphology and size distribution are not homogenous throughout the whole batch they may not be suitable as reference materials. In order to calibrate a highly sensitive machine, just like the LG-SIMS, uniformity in characteristics is the most important feature. In particular a known amount of uranium in each particle would greatly serve the purpose of a particle reference material. In the case of SIMS analysis, the production process also needs to ensure that the isotopic composition is uniformly distributed throughout the whole batch.

In light of the current techniques available for particle analysis in safeguards, reference particles should be uniform in

- uranium (respective plutonium) content,
- isotope ratios,
- size,
- chemical composition,
- morphology.

An additional requirement besides uniformity is the possibility to tailor the properties listed above so that these reference particles can provide quality assurance and quality control for the various analytical procedures. Currently it is envisioned to produce particles with various isotopic signatures.

At first, particles only containing uranium shall be produced until the production process is fully understood and under good control. Different enrichment levels are preferred, especially from 0.5, 5, <20% $^{235}$U enrichment with different isotopic compositions. At first, the production process will use certified reference materials from different suppliers like NIST, IRMM, CEA, etc. available in different forms as precursor material; mostly these are nitrate solutions. In natural uranium the isotopes $^{235}$U and $^{238}$U are the major components. Then again there are the less abundant isotopes like $^{234}$U and $^{236}$U which are called minor isotopes. In nuclear safeguards analytic the measurement of the major ratio of $^{235}$U/$^{238}$U is of key interest. Since the implementation of the new LG-SIMS, which is able to reduce isobaric interferences much more effectively, the measurement of minor ratios has become routine. The $^{236}$U/$^{238}$U ratio provides additional information. This ratio covers a wide range from $10^{-11}$ for natural uranium (NU) up to $10^{-2}$ for enriched uranium and is therefore a strong indicator in nuclear safeguards and non-proliferation applications [8]. Nowadays, the variety of RM suitable for these specific nuclear safeguards analysis methods is very limited, especially those with tailor-made isotopic content of the minor isotopes. Another important parameter for safeguards applications is the enrichment in $^{235}$U (0.5-20%, or even higher) and on the minor isotopes of uranium, plutonium and its fission products. Another field of application is its use as a reference material for age-determination. Age dating offers valuable information about the collected materials; it may even provide insight information into its composition, last date of purification and enrichment history. Calculating the decay of the radioactive nuclides provides the authorities with the ability to calculate the production date or age [9] [10]. So far, there are no reference materials available for particle age-dating methods.

The aim of this project is the “in-house” production of monodisperse particles which can be a used as RM or certified RM (CRM) eventually, especially for mass spectrometry like SIMS, LA-MC-ICP-MS, TIMS, etc. Within the next 12 months we are expecting to be able to produce monodisperse particles with reproducible results.

## 3. Particle Production Process

Production of solid aerosol particles is a two-step process. First, a stream of identical droplets is generated as a solution passes through a vibrating orifice, and second, drying, calcination of the droplets and collection of dry particles.

The generation of monodisperse particles is based on following principle: A syringe pump feeds a liquid solution through a small orifice at a predetermined and constant rate. A piezoelectric ceramic driven by an oscillating voltage potential causes the orifice to vibrate at a constant frequency, producing a uniform droplet stream. The droplet stream is introduced into the center of a turbulent air jet, dispersing droplets and preventing coagulation. The dispersed droplets mix with a larger volume of clean, dry air, which evaporates any volatile portion of the droplets.

Because each disturbance cycle of the orifice produces only one droplet, the precise size of the droplet can be deduced from the operating parameters. The two key parameters determining the droplet diameter are liquid feed rate (f) and oscillating frequency (ν).

$$\varnothing_{Droplet} = \left(\frac{6*f}{\pi*\nu}\right)^{1/3} \qquad (1)$$

For each orifice an optimal frequency range with a compatible liquid feed rate has to be defined in which it produces monodisperse aerosol particles. This frequency range and liquid feed range is restricted, because only within a certain range of feed rate and vibrating frequency the generator can uphold a steady droplet stream. Control of droplet diameter via the parameters $f, \nu$ is also limited, because of the power law of eq. (1). Effectively, the initial volume of a droplet formed by the aerosol generator is dominated by the diameter of the orifice. As a droplet consists of a volatile (solvent) and a non-volatile (solute) component its volume changes rapidly after formation. If a portion of the primary droplet is volatile, the final particle size depends on the volumetric concentration (c) of the non-volatile portion.

The particle diameter can be calculated using equation (2). [11]

$$\varnothing_{Particle} = \left(\frac{I}{C} + 1\right)^{1/3} * \varnothing_{Droplet} \qquad (2)$$

c = Volumetric amount of the soluble part of the solution
I = Correctional-term for all the non-volatile impurities

Therefore the main parameter for controlling particle size is through the concentration of non-volatile components in the feed solution. After drying and dispersing the particles are treated thermally to convert them from dried metal nitrates to the corresponding oxides. Only then we expect the particles to be stable, i.e. to maintain their integrity which is necessary to conserve the properties across the batch and during extended periods of time. It is estimated from previous experience that an additional loss of 80 Vol.-% for the calcination is realistic.

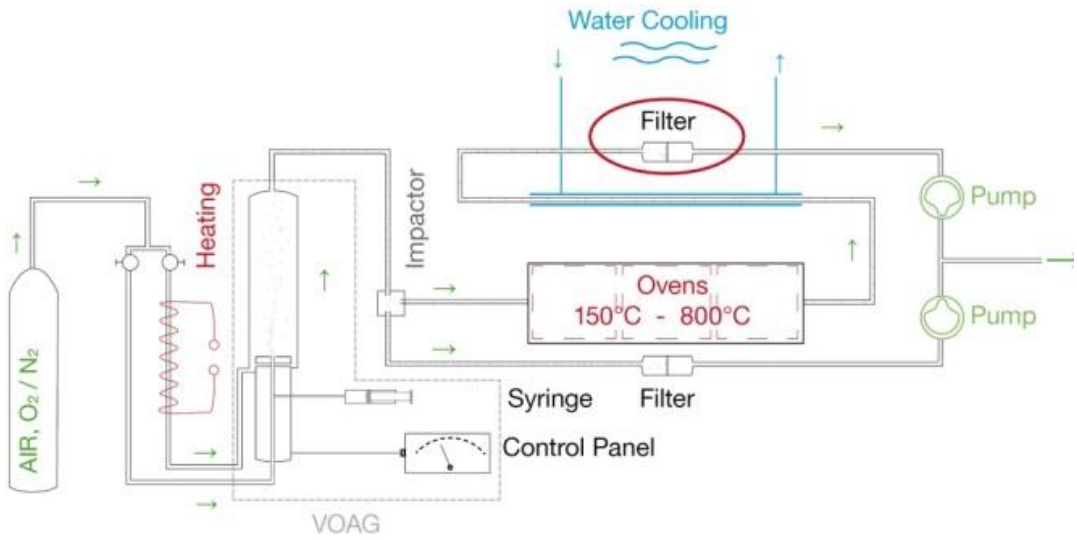


**Image 1:** Layout of the setup.

The heat treatment is accomplished 'in-stream', i.e. by heating the carrier air stream in which the particles are immersed. The calcination of the particles is performed by passing the particles through a four-zone furnace where temperatures gradually rise from 150°C to 800°C converting the particles to their oxides.
The chemical form of the final particles is determined from the feed solution and the heat treatment. Starting from certified uranium oxides (mostly $U_3O_8$) diluted aerosol solutions are prepared by dissolving the uranium oxides with nitric acid – see equation (3). After diluting the solution to a concentration of a few 100 µg U/g an aliquot is combined with an alcohol and water mixture– in our case a 1:1 ratio of ethanol and water.

$$U_3O_8 + 12\ HNO_3 \rightarrow 3\ UO_2(NO_3)_2 \cdot 6\ H_2O + 6\ NO_2 \qquad (3)$$

Thermal treatment leads to degradation of the nitrates to the corresponding nitrous-gases, mostly $N_2O_5$ and NO – see equation (4).

$$UO_2(NO_3)_2 \cdot 6\ H_2O + \Delta \rightarrow UO_3 + NO_x \qquad (4)$$

After calcination the particles have to be cooled down near room-temperature and are collected on Nuclepore track-etched polycarbonate-membrane filters (Whatman™", diameter of 25 mm and an

average pore-size of 0.6 µm). Dilution and dispersion air-streams of the system are separately controlled by membrane pumps. Both air-streams are filtered to remove the majority of particles before entering the membrane pumps in order to reduce the amount of particle contamination of equipment. A schematic diagram of the experimental setup is shown in Figure 1.

## 4. Results and Discussion of the First Test-Runs

Test runs were performed, in order to demonstrate that it is possible to produce monodisperse particles of specific diameter (e.g. 1 µm). In addition, the question of reproducibility was studied, which is of main importance because the amount of uranium per particle has to be under control. This can only be achieved if parameters like the density, geometry/diameter remains stable throughout production of the batch. The aim of these measurements was therefore the assessment of three important parameters: evaluating the morphology with SEM-measurements, amount of particles being produced and the distribution of those particles over an area of 500x500 µm.

### 4.1. 'Cold-Test' with Nd Particles

Based on several test experiments the initial parameters to operate the VOAG have been evaluated and are shown in Table 1. For test production runs using neodymium as a substitute for uranium, the generated particles were collected on standard glass, high-grade steel planchets as well as Nuclepore-filters.

| | |
|---|---|
| Dispersion stream | 0.012 L/min |
| Dilution stream | 30 L/min |
| Frequency | 72.34 kHz |
| Run-time | 2 & 5 min |
| Concentration | 0.84 mmol/L |
| Liquid-Feed Rate | $1.39 \cdot 10^{-4}$ L/min |
| Syringe Pump Run Speed | $4.2 \cdot 10^{-4}$ cm/s |

**Table 1:** Parameters for the generation of the neodymium particles.

As an aerosol solution a 1:1 mixture of high purity water and high purity ethanol (EtOH; Merck, analytical grade) was used and the concentration of neodymium(III)nitrate hexahydrate (99.998 % $Nd(NO_3)_3 \cdot 6H_2O$, Alfa Aesar, Germany) of the mixture was adjusted to 0.84 mmol/L. To collect the calcinated particles track-etched polycarbonate membrane filters were used (Whatman™, Nuclepore filters with 0.6 µm pore size and a diameter of 25 mm).
An Aerodynamic Particle Spectrometer (APS) and Optical Particle Sizer (OPS) by TSI [1] were used to determine the size of the generated neodymium particles. The APS and OPS measurements were performed online during the production test run.

### 4.2. Scanning Electron Microscopy (SEM)

Two different SEM instruments were used to characterize the morphology of the neodymium particles. In Juelich a FEI Quanta 200 scanning electron microscope (SEM) was used to determine the particle morphology, geometry, size and average diameter. The SEM is also equipped with three separate detectors, a secondary electron detector (SE), an energy-dispersive X-Ray detector (EDX) and a backscatter electron detector (BSE). Particles collected on glass plates and Nuclepore filters were measured directly. A voltage of 20.0 kV was used to record the images. The elemental composition of single particles was measured by its specific X-Ray spectrum.
In Seibersdorf a Philips XL30 with WDX 600 and an Oxford Microscope and an EDAX was used. This SEM was operated at reduced pressures ($P = 1.1 \cdot 10^{-5}$ Pa) and a very low accelerating voltage of V= 9kV. The results were compared with known peaks of the following elements: Fe (Kα), Si (Kα), Cr (Kα), Fe (Kβ), C (Kα), O (Kα), Nd (Kα) und Nd (Kβ).

[1] http://www.tsi.com/Particle-Sizers/

## 4.3. Size and Morphology of Nd Test-Particles

Online and offline measurements of the generated neodymium oxide particles using the OPS, APS and SEM were in general agreement indicating a typical size of the particles of 4.2 - 5.0 µm. This is in good agreement with the calculated size using equation (2). A difference in the average diameter of about 25% is suggested by the results obtained from the APS and OPS measurements. SEM measurements confirmed the accuracy of the APS measurements, which can be performed online during production runs. APS provides an opportunity to measure the diameter online and with high accuracy. The average diameter of the neodymium particles of 4.7 µm is confirmed on the measurement of ten random particles by SEM. The standard deviation of the average particle diameter is less than 5% indicating an acceptable uniformity of the generated particles.

The SEM measurements showed that all particles were spherical in shape except those collected on the Nuclepore-filter membranes. Latter were not only deformed but also slightly smaller in size. All particles showed a coronal-like pattern around their impact site on the collection medium. This indicates that the particles were not completely dry before impacting on the target. Figure 3 shows the geometry and morphology of two particles collected on different substrates. The left image shows a wet neodymium particle impacted on a glass-planchet with a very distinctive splash-pattern, and the right image a particle collected on a stainless steel target.

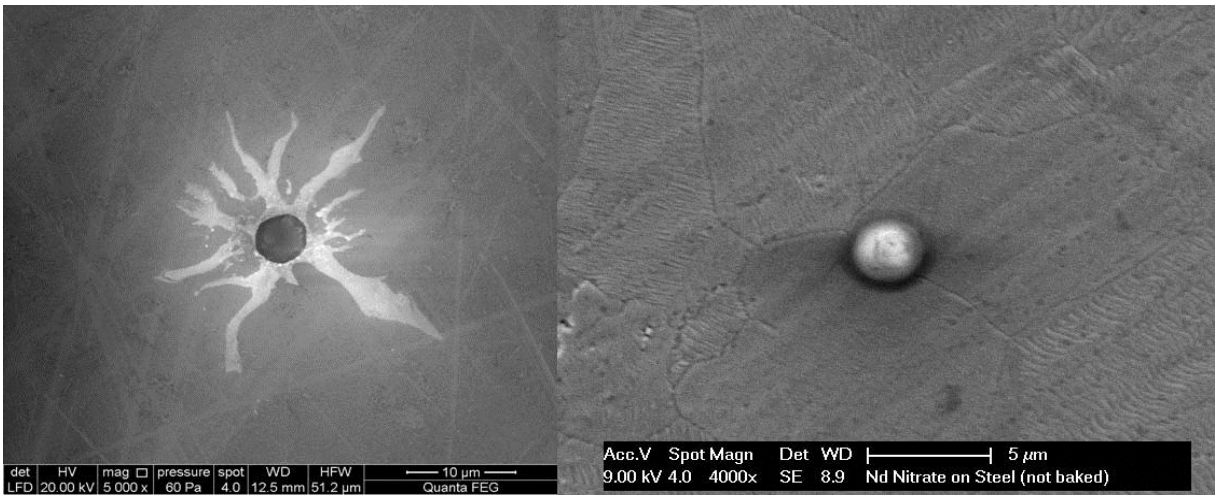


**Figure 3:** Left: $Nd(NO_3)_3 \cdot 6H_2O$ sphere on a glass-target and Right: $Nd(NO_3)_3 \cdot 6H_2O$ sphere on a steel-target.

Online measurements with the APS and OPS were also used to measure the amount of particles being produced. These measurements were performed right before impacting on the specific targets. The results can be seen in table 2 – they were achieved with the parameters of table 1. The results also show a big discrepancy between the amount of particles before the Nuclepore Filter and the glass and steel targets. This is due to the fact, that the Nuclepore Filter is integrated into the sample unit which is integrated in the dilution air stream, where the glass and steel targets on the other hand were just damped with a hose. This clearly indicates the effects of different geometries to the air currents inside the apparatus. It has to be noted that different volumes of the dilution air stream had a huge impact on the amount of particles being detected.

| | |
|---|---|
| Before glass filter | 600 particles/min |
| Before steel filter | 620 particles/min |
| Before Nuclepore filter | 200 particles/min |

**Table 2:** Particle density before targets measured with the APS.

One of the first conclusions could be drawn from the online measurements with the APS and OPS. We were able to measure the amount of particles reaching the targets and could compare those numbers with the amount of particles being produced. Clearly, it showed that the dependency of the internal air

currents (both dilution and dispersion air) to small differences to the setup as well as to some parameters like the liquid feed rate and the vibration frequency of the orifice - see chapter 3 last paragraph. Clear indications for those effects are the values recorded in table 2; they just depict the optimal values being recorded. SEM images taken over an area of 500x500 µm could also confirm that the amount of particles reaching the target varied drastically depending on changes to the frequency of the orifice vibration as well as to changes to the sampling-setup.

Another obvious difference could be observed between the particles which were damped on the targets and those which impacted on the Nuclepore filters. Particles collected on glass and steel-planchets were less skewed when compared with those impacted directly on Nuclepore filters. SEM images also suggest that most particles were not completely dry prior to impacting. This issue will be addressed by using preheated air in the dispersion air stream

SEM images over an area of 500x500 µm were made to investigate the distribution of particles over a larger area. It could be observed that very few agglomerations occurred and that all particles were had uniform size.

## 5. Conclusion

A Vibrating Orifice Aerosol Generator was installed and preliminary studies on the production of uniform particles were carried out. SEM images of the particles confirmed a well-defined size, geometry and production of discrete particles with a very low degree of coagulation.

Further optimization of the experimental set-up and adjustments of the experimental parameters will be performed over the next few months. First runs with nuclear materials are planned in the second half of 2013, with the main objective of “in-house” production of monodisperse particles which can be a used as Quality Control materials for particle analytical techniques such as SIMS, LA-MC-ICP-MS, and TIMS.

## 6. Acknowledgements

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# Unattended Safeguards Instrumentation at Centrifuge Enrichment Plants

**L. Eric Smith, Alain R. Lebrun, Rocco Labella**

International Atomic Energy Agency
Wagramerstrasse 5, Vienna, Austria
E-mail: e.smith@iaea.org, a.lebrun@iaea.org, r.labella@iaea.org

***Abstract:***

*As global uranium enrichment capacity under international safeguards expands, the International Atomic Energy Agency (IAEA) is challenged to develop effective safeguards approaches at gaseous centrifuge enrichment plants, particularly high-capacity plants, while working within budgetary constraints. New safeguards approaches should meet the high-level verification objectives for such facilities (i.e., timely detection of: diversion of declared material, excess production beyond declared amounts, and production of enrichment levels higher than declared), but should also strive for efficiency advantages in implementation, for both the IAEA and operators. Under the Agency's State-level approach to safeguards implementation, the Agency needs a flexible toolbox of technologies, allowing tailoring of safeguards measures for each individual enrichment facility. In this paper, the potential roles and development status for three different types of unattended measurement instrumentation are discussed. On-Line Enrichment Monitors (OLEM) could provide continuous enrichment measurement for 100% of the declared gas flowing through unit header pipes. Unattended Cylinder Verification Stations (UCVS) could provide unattended verification of the declared uranium mass and enrichment of 100% of the cylinders moving through the plant, but also apply and verify an 'NDA Fingerprint' to preserve verification knowledge on the contents of each cylinder throughout its life in the facility. Sharing of the operator's load cell signals from feed and withdrawal stations could count all cylinders introduced to the process and provide periodic monitoring of the uranium mass balance for in-process material. The integration of load cell, OLEM and UCVS data streams offers the possibility for 100% verification of declared cylinder flow, and enables the periodic verification of the declared $^{235}U$ mass balance in the plant. These new capabilities would enhance the IAEA's effectiveness in meeting the high-level verification objectives at enrichment facilities. In addition, such unattended instrumentation could reduce or eliminate the need for routine interim inspections, and significantly reduce the need for drawing samples from gas and cylinders during inspections—thereby achieving operational efficiencies.*

**Keywords:** enrichment plants; unattended instruments; non-destructive assay

## 1. Introduction

The IAEA's model safeguards approach for gas centrifuge enrichment plants [1] describes the challenges associated with safeguarding large centrifuge enrichment plants, and defines the high-level verification objectives for enrichment plant safeguards approaches, i.e., the timely detection and deterrence of:

a) diversion of natural, depleted or low-enriched $UF_6$ from the declared flow in the plant;
b) misuse of the facility to produce undeclared product (at the normal product enrichment levels) from undeclared feed (i.e., excess production);
c) misuse of the facility to produce $UF_6$ at enrichments higher than the declared maximum, in particular highly enriched uranium.

At present, the IAEA's safeguards approaches at enrichment plants are based on a combination of routine and random inspections, during which time a number of verification activities are performed, including: environmental sampling (ES) for subsequent laboratory analysis; collection of $UF_6$ samples from in-process material and selected cylinders for subsequent destructive analysis (DA) in a laboratory; weighing and nondestructive assay (NDA) of a subset of the plant's cylinder flow and inventory. The weight measurements of cylinders are performed using either operator-owned scales or the IAEA's portable hanging load cells, while the NDA measurements utilize handheld gamma-ray spectrometers combined with ultrasonic wall-thickness gauges.

Detection of prominent diversion scenarios could be improved at enrichment plants if the IAEA could monitor 100% of material flows and periodically calculate independent uranium and $^{235}U$ mass balances for the facility. However, human and financial resources preclude continuous inspector presence at the facility to measure all of the material flow using today's attended methods. Further, the portable measurement methods currently used by inspectors have relatively low accuracy for both the total uranium mass and $^{235}U$ enrichment in a cylinder, which would lead to very large uncertainties on a $^{235}U$ balance based on such instruments. The poor accuracy of today's cylinder verification instruments necessitates additional safeguards measures, including the destructive analysis of $UF_6$ samples drawn from some of the cylinder population. These are among the reasons that the IAEA is exploring how unattended instruments capable of continuously and more accurately verifying material flows (both in-process gas and cylinders) on a quasi-continuous basis could help improve the deterrence and timely detection of protracted diversion scenarios. This paper discusses three candidate unattended measurement systems currently being considered by the IAEA: On-Line Enrichment Monitors (OLEM), Unattended Cylinder Verification Stations (UCVS), and sharing of the operator's load cell signals (Figure 1).

These unattended instruments are potential tools in a flexible toolbox of safeguards measures that is aimed at addressing the verification challenges posed by advanced centrifuge technologies and the growth in separative work unit capacity at modern centrifuge enrichment plants [2][3][4]. Permanently installed, unattended instruments could perform the routine and repetitive measurements previously performed by inspectors, thereby allowing the inspectors to use their time on tasks and investigation that depend more heavily on human intuition and decision making. When combined with other safeguards measures, unattended instruments at centrifuge enrichment plants have the potential to significantly improve the Agency's effectiveness to detect and deter the primary diversion scenarios of concern, while simultaneously improving the efficiency of facility-level safeguards approaches. Further, the unattended measurement systems have the potential to be beneficial to facility operators as well, for example for process control, for meeting regional or State regulatory requirements, or to ease and expedite the process for releasing cylinders from the facility. Identifying and developing improvements in safeguards efficiency, while maintaining or improving effectiveness, are important considerations as the IAEA fully implements the State-level concept and evolves the role of safeguards technologies [5].

To support the discussions that follow, a reference centrifuge enrichment plant is defined. This plant represents the modern, large-capacity centrifuge facilities that are a primary motivation for the IAEA's study of a new generation of safeguards measures and approaches. The reference facility is 4,000 tonne SWU/year, with 8 process units consisting of 10 cascades each, and utilizing $UF_6$ withdrawal by desublimation directly into product and tails cylinders. Two material balance areas (MBAs) are defined in the plant: a process MBA and a storage MBA. A schematic overview of this reference facility is given in Figure 1, and a summary of the roles of each unattended instrument is given here.

The OLEM could provide continuous measurement of 100% of the declared gas flowing through unit header pipes, a key capability for the detection of the higher-than declared production and diversion-from-declared scenarios. In the reference large-capacity enrichment plant described above, 16-24 OLEM units would be required, depending on whether the feed is monitored. The OLEM units would be owned and operated by the IAEA, but include data-security provisions to allow sharing with the operator (e.g. for process control and criticality control purposes) or other stakeholders (e.g. regional or national authorities). Sharing of the operator's load-cell signals from feed and withdrawal stations has the potential to count cylinders introduced to the process and to provide periodic balance of the uranium mass for the in-process material at the plant. Load-cell monitoring supports the detection and deterrence of excess production scenarios in a way that other unattended instrumentation cannot. More than 100 feed or withdrawal stations would need to be monitored in the reference facility. UCVS units could provide unattended verification of the declared uranium mass and enrichment in 100% of

the cylinders moving through the plant, but also apply and verify an 'NDA Fingerprint' to preserve verification knowledge on the contents of each cylinder throughout its life in the facility, without the need for an inspector's presence to apply and verify traditional seals. The UCVS NDA features also have the potential to provide independent cross-verification of the signals from operator weighing systems. The UCVS would be built around the operator's accountancy scales, so that two or three UCVS units might be utilized in each plant. Apart from the accountancy scales, UCVS would be owned and operated by the IAEA, but include data-security provisions so that data streams could be shared with the operator (e.g. for cylinder tracking and process control).

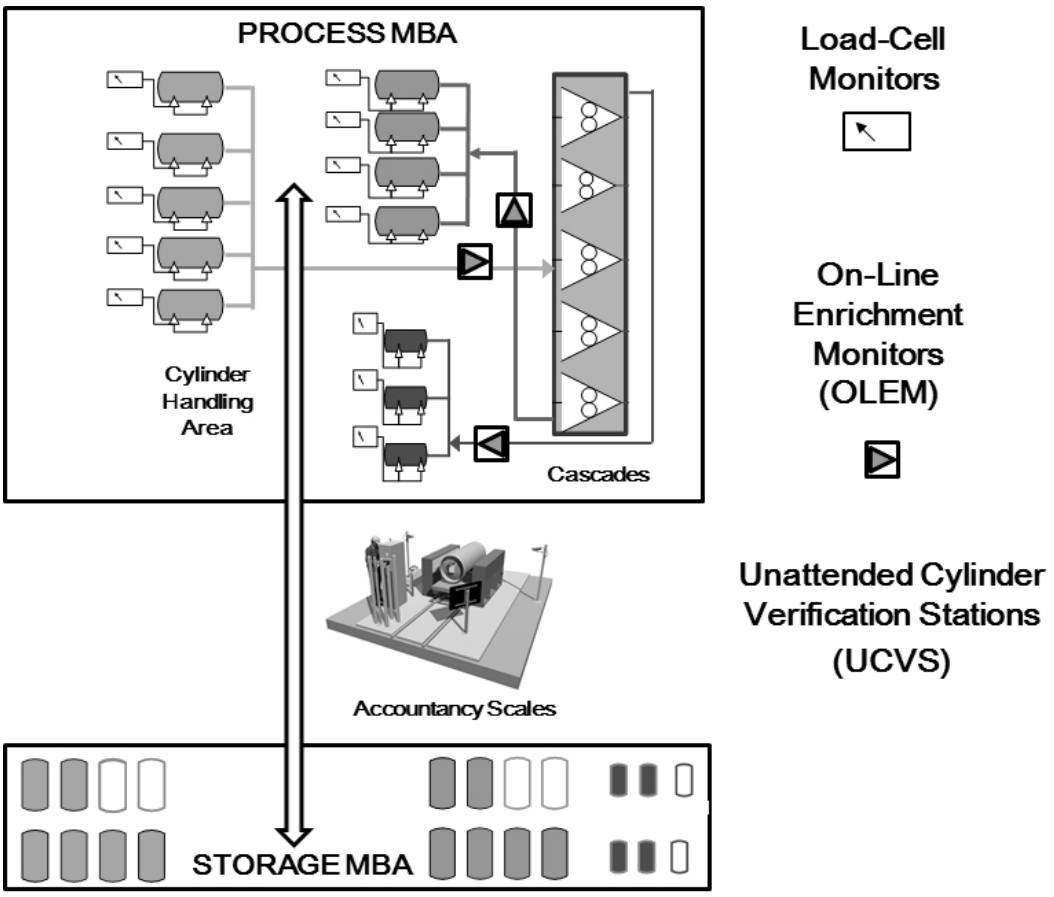


**Figure 1.** Schematic overview of load-cell monitors, OLEM and UCVS in an enrichment facility divided into a process material balance area (MBA) and a storage MBA.

A soon-to-be-published journal article from the IAEA provides detailed discussion of the potential roles, development status and remaining development questions for these three candidate unattended measurement systems [6]. An example case study in that paper demonstrates quantitatively how unattended instruments could simultaneously improve effectiveness and efficiency over today's measures, through the integration of load cell, OLEM and UCVS data streams that can support 100% verification of declared cylinder flow and enable the periodic verification of the declared $^{235}$U mass balance in the plant. The case study illustrates how the continuous presence and relatively high accuracy of the OLEM and UCVS $^{235}$U assay could support the detection of protracted diversion scenarios in a way that has never before been viable for the IAEA, due to accuracy and operational limitations associated with portable instruments for cylinder assay.

This paper draws from [6] to provide an overview of the IAEA's evolving vision for a new generation of unattended safeguards instruments at enrichment plants, and how those instruments might support a flexible toolbox of verification measures that the IAEA could draw upon under its State-level approach to safeguards implementation.

## 2. Unattended Instrumentation in Context of the IAEA's State-Level Concept

Under the IAEA's State-level concept for defining safeguards approaches, the specific measures implemented at each facility will depend on a set of factors that include: State-specific characteristics (e.g. additional protocol in force); effectiveness in detecting and deterring the key diversion scenarios for that facility; plant design (e.g. size, re-configurability); operator acceptance (e.g. proprietary concerns); and efficiency (e.g. cost, complexity of safeguards measures) [2]. In order to optimize the efficacy and efficiency of safeguards measures at each different enrichment facility under safeguards, the IAEA needs a flexible toolbox of technologies (e.g. unattended and attended) and inspection options (e.g. announced and unannounced).

The IAEA's guiding philosophy is to rely on unattended instruments to perform routine, repetitive measurements, thereby unencumbering inspectors to do the investigative activities that rely on human intuition, tacit knowledge and decision making, such as design information verification or verifying the absence of indicators for undeclared activities. This implementation philosophy should lead to important operational advantages for all stakeholders, for example a significant reduction or elimination of routine interim inspections, reduction of material sampling activities during inspections, and the expediting of product-cylinder release for the operators.

Figure 2 provides a graphical representation of how the IAEA's inspection and technology tools (left side of fulcrum) might be balanced against the data needed to draw safeguards conclusions at the facility level (right side of fulcrum). For the tools on the left side of the fulcrum, the level of data independence increases from right to left. The potential volume of data derived from each measure during a given material balance period is depicted by the height of the box for each tool. For example, the volume of data derived from operator-owned and maintained load cells and mass spectrometers (boxes with grey hash) may be quite large, but this less-independent data would exert less force on the fulcrum arm than the same volume of more-independent data coming from IAEA-owned-and-operated instruments such as OLEM and UCVS. It is important to note that unattended instrumentation technology would always be accompanied and complemented by inspections (e.g. announced inspections, unannounced inspections and periodic physical inventories) that include investigative and data security activities.

As an example of how the State-level concept might be applied to enrichment plant safeguards, consider 'Facility Y' located in a State with a comprehensive safeguards agreement and additional protocol in force, where integrated safeguards (e.g. including results from complementary access and open-source information analysis) has supported the broader conclusion that there are no undeclared nuclear material or activities in the State. Under these conditions, the importance of the excess production diversion scenario would be reduced, since the IAEA would have already concluded that there are no undeclared enrichment plants to further enrich undeclared low-enriched uranium product diverted from Facility Y. The measures implemented at Facility Y therefore, would be focused on detecting the higher-than declared production scenario, and verifying the declared material flows, as efficiently as possible. The safeguards measures implemented at this Facility Y are likely to be relatively limited compared to the measures implemented at Facility X, which is located in a State without an additional protocol in force, and where the provision of safeguards-relevant information to the IAEA has been more limited. For Facility X, the excess production scenario is of high importance because the IAEA is not able to draw the broader conclusion regarding undeclared enrichment facilities. In Facility X, the data needed to draw safeguards conclusions (right side of fulcrum in Figure 2) will likely require a relatively 'heavy' suite of safeguards measures (left side of fulcrum).

Facilities X and Y are fictitious, created only to provide tangible examples of how the IAEA might define facility-specific safeguards measures under the State-level concept, using a flexible toolbox of unattended instruments and inspection authorities. In the remainder of this paper, more details and discussion about each of the individual unattended instruments are provided, along with more thorough descriptions of how the data streams from such instruments might be integrated at the facility level. Facility X is used as an illustrative example in these discussions.

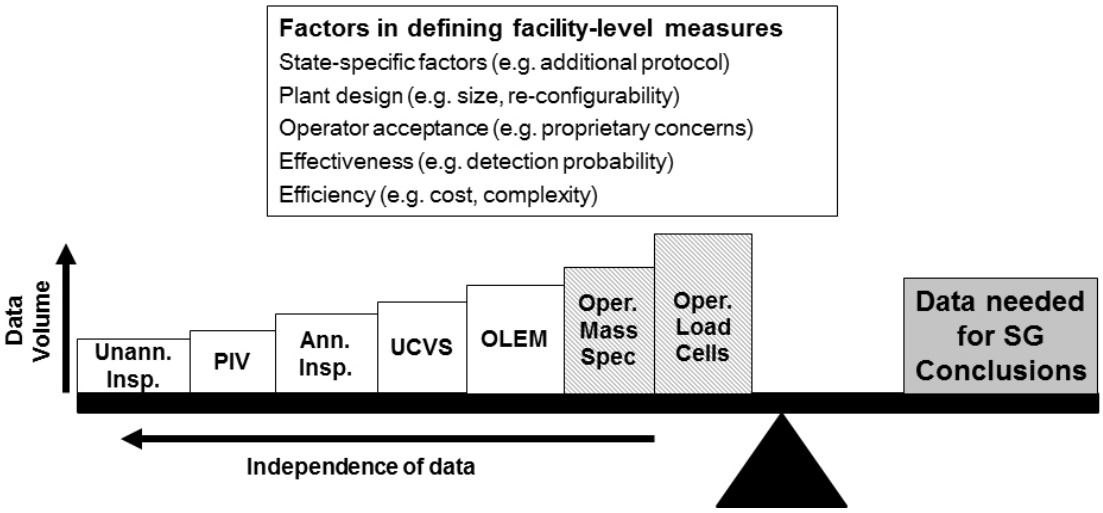


**Figure 2.** Depiction of how the IAEA might balance a toolbox of safeguards measures (left side of fulcrum) including announced and unannounced inspection activities, a physical inventory verification (PIV) and unattended instrumentation, against the data needed to draw facility-level safeguards conclusions (right side of fulcrum). Only selected measures on the left side of would be implemented at a given facility, depending on the data needed to draw safeguards conclusions under the State-level concept.

## 3. Roles for Unattended Instrumentation at Facility X

The safeguards measures at Facility X include substantial utilization of unattended systems—load cell monitoring combined with OLEM, and UCVS. Consequently, this facility provides a convenient example for discussing how the data streams from these instruments could be integrated to allow the inspectorate to address the three relevant diversion scenarios: 1) diversion from declared, 2) excess production, and 3) higher-than-declared enrichment. It is assumed in this discussion that Facility X contains two Material Balance Areas (MBAs, see Figure 3). The Process MBA includes the cascades, feed and withdrawal stations, weighing and sampling areas, and scrap and waste recovery. The Process MBA in Facility X includes the cylinder blending stations, though it is possible that the blending area could be a separate MBA, or even within the Storage MBA, in facilities under IAEA safeguards.

The excess production scenario could be addressed by counting the cylinders introduced to the cascades to ensure that only declared cylinders are utilized, and via the continuous monitoring of the in-process $UF_6$ material balance ($MUF_{proc}(t)$ in Figure 3). This material balance would be based on the measured feed, product and tails mass flow rates (*F, P, T* respectively) in each enrichment unit (each of which might consist of 8-10 cascades), as determined from the sharing of operator load cell data from all of the feed and withdrawal stations in the unit. The time-dependent mass data, *M(t)* from the operator's load cells could be shared with the IAEA to determine the time periods during which specific cylinders are being filled (for product and tails stations) or are being withdrawn as plant feed. The material unaccounted for (*MUF*), would be calculated by the IAEA at time intervals negotiated with the operator, taking into consideration for example, the protection of operator's proprietary information. Under normal operation, the *MUF(t)* for total uranium calculated by the IAEA's sharing of the operator's load cells would be expected to be relatively small over short material balance periods, and consistent with mass decrements that are typical of normal operation for the plant (e.g. due to sampling, scrap, holdup). Unattended monitoring of the feed and withdrawal stations could also help to streamline inspection activities (e.g. to minimize cylinder switchover activities).

The OLEMs on each unit header pipe would continuously measure the time-dependent relative uranium enrichment, *E(t),* in weight percent $^{235}U$, of the gas filling or the gas being withdrawn from the cylinders. *E(t)* could be used in several ways. First, it could be combined with the *F, P* and *T* total uranium mass flow rates recorded by feed and withdrawal station load cells, to calculate *MUF(t)* for $^{235}U$. The IAEA then, could monitor for the excess production scenario using both the uranium and $^{235}U$ mass balances on the in-process gas.

OLEM data could also be used to calculate the average enrichment of the $UF_6$ in cylinders, $E_{cyl}$, by weighting the *E(t)* data for each cylinder time window by the *M(t)* for that same time window. By coupling the load cells and OLEMs in this way, a high-accuracy, independent measurement of $E_{cyl}$ is produced. Alternatively, in cases where the sharing of load-cell signals is not acceptable or practicable, less-direct approaches to deriving mass-flow data could be considered. Such approaches may be viable, for example, in modern enrichment plants where the product enrichment level in each unit header is typically held as stable as possible for relatively long periods of time (e.g. 4.42% for several months). Under these assumptions, *E(t)* = constant = $E_{cyl}$ . *M(t)* could be assumed constant, or inferred from other plant variables.

Another important role of the OLEM units is the continuous monitoring of in-process gas for early detection of greater-than-declared enrichment levels. Because of the location of the OLEMs (see next section), a scenario involving cascade recycle and early takeoff inside the cascade halls is not precluded, but such a scenario is likely to require the operator to make undeclared facility modifications that would be prone to identification during unannounced inspections. Load-cell monitoring and other IAEA tools (e.g. environmental sampling during unannounced inspections) could also be used to address such early takeoff scenarios.

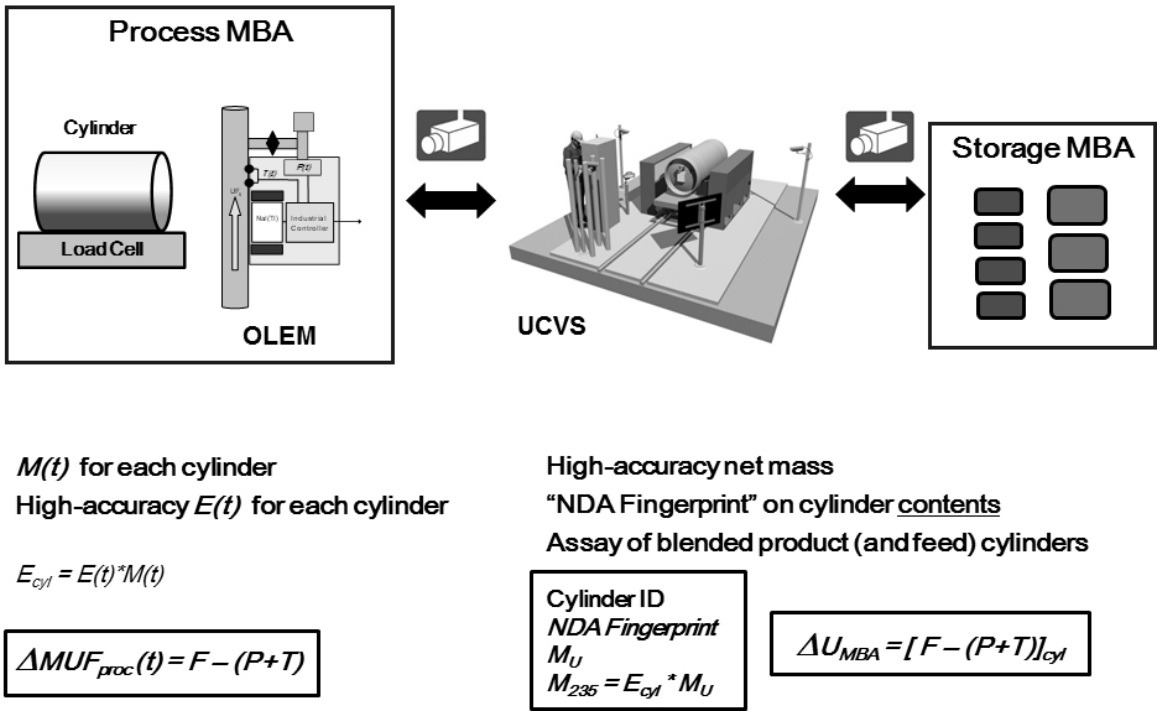


**Figure 3.** Schematic overview of how load-cell monitors, OLEM and UCVS might be integrated in a two-MBA plant similar to Facility X. Other containment and surveillance measures (e.g. cameras) in the facility would complement the unattended measurement systems.

An extremely important piece of data for the facility-level instrumentation system at Facility X is the net uranium mass, $M_U$, in each cylinder. This mass would be based on the full and tare weights measured by the operator's accountancy scales, as reported through the sharing of the accountancy scale weight tickets. The UCVS, which would be built around the operator's accountancy scales in order to leverage the cylinder characterization opportunity presented by the facility's normal cylinder weighing operations, could be the interface for the collection and utilization of the scale weight tickets. Since $M_{235} = E_{cyl} \cdot M_U$, the combination of load cell, OLEM and UCVS data allows determination of $^{235}U$ mass for each unblended cylinder (blended cylinders are discussed later). Further, the accountancy scale weight values can be important as a confidence building measure for the less-accurate tare and full weights reported from the load cells at the feed and withdrawal stations. Direct, independent assay of $M_U$ using UCVS radiation signatures might also be a confidence-building measure on the authenticity of accountancy scale and load-cell data.

Once the values of $E_{cyl}$, $M_{235}$ and $M_U$ are established for each cylinder, it would be ideal that continuity of knowledge (CoK) on that cylinder and its contents would be maintained as long as the cylinder remains at the facility. This is a particular challenge in gaseous centrifuge enrichment plants since the traditional tool for CoK on nuclear material containers, metal or electronic seals, would require very frequent inspector presence to either emplace or remove seals; there exists no practical mechanism for unattended placement and removal of such seals. (There is a precedent for operators to either emplace or remove seals, but not both.) A new concept is needed to address this CoK challenge, and for that purpose, the concept of an 'NDA Fingerprint' applied and verified by the UCVS is being investigated by the IAEA. This NDA Fingerprint is intended to compensate for the lack of traditional, continuous CoK on the verified cylinders, by providing a means to periodically confirm, in an unattended fashion, that the contents of the cylinder are unchanged.

The NDA Fingerprint is a collection of distinguishing attributes for the cylinder contents that could include, for example, total uranium mass, $M_{235}$, various isotopic ratios (e.g. $M_{234}$ / $M_{235}$ and $M_{232}$ / $M_{235}$) and the spatial distribution of $^{235}U$ within the cylinder. The task of 'setting' and verifying the NDA Fingerprint would be performed by the UCVS. A UCVS scan would occur each time a cylinder crosses an MBA boundary, to provide periodic re-verification of the cylinder contents, until the time the cylinder is shipped offsite. The UCVS and NDA Fingerprint concept could also be extended to facilities preceding the enrichment plant (e.g. for feed cylinders from the uranium conversion facility) and following the enrichment plant (e.g. receipt of the product cylinders at fuel fabrication plants), as a part of a State-level verification approach.

The UCVS units could play other important roles in Facility X, for example in terms of cylinder identification and tracking, and for the verification of the $UF_6$ in blended cylinders for which there would be no associated OLEM-based measurement of $E_{cyl}$. Another potential benefit of the UCVS would be to ease and expedite the product cylinder release process for the operators. For example, product cylinders ready for shipment could be brought to the appropriate accountancy scale for final confirmation of $M_U$ and verification, via the NDA Fingerprint collected by the UCVS, that the $UF_6$ inside the cylinder is unchanged since the cylinder was previously measured at the boundary of the Process MBA. A conceptual overview of how unblended product cylinders could be verified and released from the facility using unattended instrumentation is given in Figure 4 below.

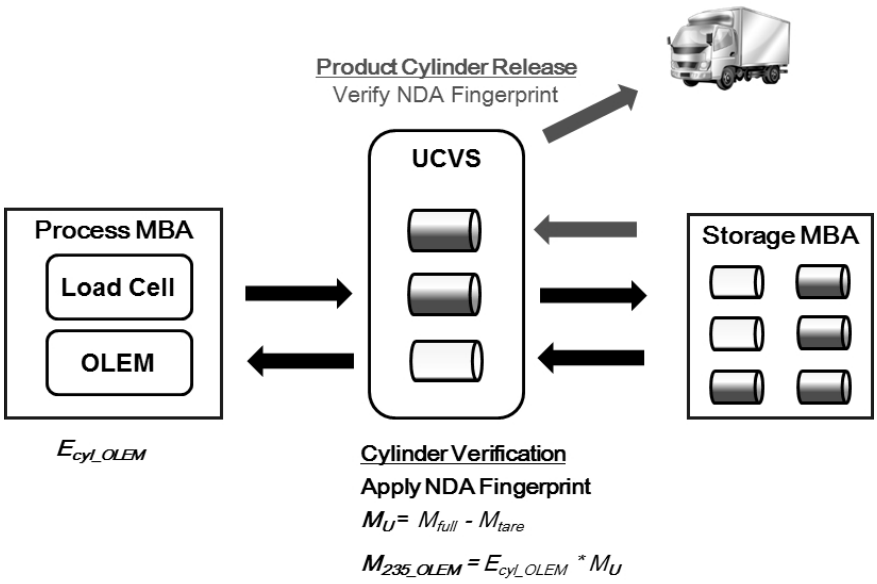


**Figure 4.** Conceptual overview of how an unblended product cylinder could be verified and released from the facility using a combination of load cell monitoring, OLEM and the UCVS. The empty cylinder would begin in the storage MBA at right, be characterized by the UCVS on its way into and out of the Process MBA. Data from load cells and OLEM ($E_{cyl\_OLEM}$) would support high-accuracy calculation of $M_{235\_OLEM}$ in each cylinder. When the operator is ready to ship the cylinder off-site (grey arrows at top), the UCVS's NDA Fingerprint capability would be used to verify the constancy of the cylinder contents since production.

## 4. Technology Development Status

Though many development challenges remain before field implementation of these technologies could be considered by the IAEA, there are encouraging results and indications coming from development efforts in the safeguards community. A brief summary of the development status for OLEM, UCVS and shared-use load cells is given here; a full discussion can be found in [6].

Modeling-based OLEM viability studies by the IAEA established the expected range of measurement uncertainties under representative plant conditions [7] and helped inform the IAEA's user requirements and performance targets for OLEM [8]. A collaborative field measurement campaign performed by Los Alamos National Laboratory (LANL) and Urenco at Urenco's Capenhurst (England) facility provided invaluable experience and empirical support for the viability of OLEM [9]. These field tests extended the community's understanding of how pressure transients can be used for wall-deposit calibration, and confirmed that measured uncertainties on product-gas enrichment are consistent with IAEA's modeling-based performance targets. Collectively, these simulation and empirical studies indicate that well-calibrated instruments on header pipes are capable of meeting OLEM performance targets: 1%, 2% and 3% (relative one-sigma) uncertainties for product, feed and tails gas streams, respectively. OLEM field prototypes are now being developed under the United States Support Program to the IAEA [10], in a collaboration between Oak Ridge National Laboratory (ORNL) and LANL.

Modeling-based viability studies for the NDA functions of the UCVS have been undertaken, and the strengths and limitations of two different cylinder assay methods are currently being studied. Pacific Northwest National Laboratory (PNNL) has developed a hybrid cylinder assay technique that utilizes an array of NaI(Tl) spectrometers to simultaneously measure the direct 186-keV signature from $^{235}$U and via high-energy gamma rays induced by neutrons in $^{56}$Fe and the NaI(Tl) itself, the total neutron emission rate from the cylinder. The 186-keV signature provides an unambiguous measure of $E_{cyl}$. Under assumptions of known $^{234}$U/$^{235}$U behavior in the plant, the total neutron signal can be calibrated to total $M_{235}$ in the cylinder [11][12]. Over several field campaigns using PNNL's Hybrid Enrichment Verification Array (HEVA) prototype for the assay of Type 30B cylinders with enrichments ranging from 2.0% to 5.0%, relative uncertainties of approximately 3% for $E_{cyl}$ and 4% for $M_{235}$ were reported. LANL's Passive Neutron Enrichment Monitor (PNEM) employs moderated $^{3}$He modules to measure the singles and doubles neutron emission rates from the cylinder [13][14]. The singles emissions come primarily from the $^{234}$U, which under an assumption of known $^{234}$U/$^{235}$U behavior allows determination of $^{235}$U mass. The singles to doubles ratio allows calculation of the cylinder enrichment level. A field campaign using a PNEM prototype included the assay of Type 30B cylinders with enrichments ranging from 2.0% to 5.0%, and measurement times of approximately 20 minutes, but relative uncertainties for $E_{cyl}$ and $M_{235}$ over the measured cylinder population have not been published. The European Commission's Joint Research Centre (JRC) at Ispra provided both qualitative and quantitative assessments of the systematic and statistical uncertainties that arise when utilizing various radiation signatures to assay cylinder contents [15]. While this prior work on NDA methods for UCVS has been encouraging, the expected measurement uncertainties in realistic enrichment plant operation are not yet fully understood, nor has the viability of the NDA Fingerprint, or the direct assay of total uranium mass, been explored.

Though there are no new developments needed in terms of the operator's load cell systems themselves, there remain a number of unanswered questions about the statistical and systematic uncertainties associated with data streams from those load cells, how those uncertainties propagate through a facility-level mass balance, and the effectiveness for detecting excess production scenarios using a mass balance with those uncertainties [6][16]. Work is also underway to develop and assess hardware and software approaches for data sharing that meet IAEA requirements for data security. Ongoing studies by the European Commission's Joint Research Centre (JRC) at Ispra, in an operating enrichment plant in France, will aid the community's understanding of hardware, data collection methods, and operator tolerance for the sharing of near-continuously produced process control data that is often considered proprietary [17]. Innovative techniques for building confidence in the integrity and completeness of the load-cell data are being considered by ORNL [18].

## 5. Conclusions

Key themes and conclusions from discussed in [6] and summarized in this paper include:

**Potential for unattended instruments to substantially improve effectiveness and efficiency:** Creative integration of unattended instrumentation and coupling to unannounced inspections has the potential to achieve significant improvements in the timely detection and deterrence of enrichment-plant diversion scenarios. Significant improvements in cylinder assay accuracy offered by the integration of load-cell monitoring, OLEM and UCVS opens the possibility of near-real-time verification of the declared $^{235}U$ mass balance at the perimeter of the process MBA. This capability was never before available to the IAEA, due to the limited sampling of cylinders during interim inspections and the relatively poor measurement accuracy of the portable devices used for cylinder verification measurements. Such unattended instrumentation combinations would also lead to substantial efficiency improvements, for example the elimination or significant reduction of routine interim inspections, and a reduction in sampling of gas and cylinders during inspections.

**Need to characterize and quantify instrument measurement uncertainties:** A solid understanding of the achievable measurement uncertainties in realistic plant environs, and the corresponding uncertainty budgets (i.e. the relative contributions of random and systematic errors), for each of the candidate unattended technologies is invaluable to the IAEA to help guide and refine the development of user requirements, and also to support the analysis of facility-level diversion scenarios. These instrument-level uncertainty studies are more advanced for OLEM and UCVS than for load-cell monitoring, but further investigation is needed for all three unattended instruments.

**Need to characterize potential vulnerabilities and data security challenges:** Instrument development activities should address, at least in the preliminary sense, key vulnerabilities or spoofing possibilities, since vulnerabilities could ultimately define whether the technology could be adopted by the IAEA. The same could be said for data security measures and whether those measures are sufficient to meet IAEA's requirement to draw independent safeguards conclusions, while at the same time facilitating the sharing of instrument data with operators and other stakeholders.

**Need to identify and pursue long-term field testing opportunities:** In order to build confidence in the lifecycle viability of new unattended technologies, long-term field testing in representative facilities with field prototypes meeting IAEA's user requirements is needed. Developing flexible testing agreements that include tolerance for instrument 'learning periods' and down-time for revision or troubleshooting is critical. The IAEA will continue to engage Member States and facility operators to identify suitable testing opportunities.

**Need to explore 'win-win' opportunities for IAEA and operators:** Discussions between the IAEA and operators have suggested that IAEA's unattended instruments may have notable benefits for facility operators, for example for process control, criticality safety or to meet requirements from state authorities. Such opportunities should be identified and pursued as early as possible, so that the necessary hardware and software capabilities (e.g. sensor duplication, data branching methods and data security hardware/software) can be integrated efficiently, rather than as an afterthought.

The concepts presented in this paper should not be considered a comprehensive study of all implementation options being considered by the IAEA but rather, as a starting point for discussions regarding the potential and challenges associated with the use of unattended measurement systems at enrichment plants. It remains to be seen whether any of these technologies described in this paper, or combinations thereof, will be deployed in field operations. Ultimately, deployment decisions will be based on a combination of factors that include efficacy, lifecycle cost, and operator acceptance.

## 6. Acknowledgements

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# Enhanced IAEA/EURATOM Cooperation and Reinvigoration of the NPA

**Igor Tsvetkov, Wan Sou Park, Therese Renis, Norbert Peter**

International Atomic Energy Agency (IAEA)

**Paul Meylemans, Wolfgang Kahnmeyer, Johan Dackner, Herbert Dratschmidt**

European Commission (EC)

***Abstract:***

*The performance of State and regional authorities responsible for safeguards (SRA) and their effectiveness have a significant impact upon the effectiveness and efficiency of safeguards implementation. The IAEA is considering ways to make full use of the effectiveness of SRAs and place greater reliance on their support to the IAEA in implementing safeguards, while maintaining the IAEA's ability to draw its independent safeguards conclusions. It is important that States and/or regions have competent authorities responsible for safeguards and enact legislation to enable the establishment and maintenance of effective systems of accounting for and control of nuclear material (SSAC/RSAC), including the capability to conduct verification activities independently pursuant to such legislation. The IAEA is considering conditions under which enhanced cooperation arrangements with SSACs/RSACs could be achieved to allow the IAEA to make better use of their systems for optimizing routine safeguards implementation.*

*The Safeguards Agreement between the non-nuclear-weapon States of EURATOM, EURATOM and the IAEA (INFCIRC/193) mandated a cooperative working arrangement between the IAEA and EURATOM to facilitate the implementation of safeguards and to avoid unnecessary duplication of safeguards activities. Until 1992, the established co-operation between the IAEA and EURATOM was primarily based on "observation" and "Joint Team" arrangements. A New Partnership Approach (NPA) was agreed between the IAEA and the European Commission in 1992 to strengthen safeguards collaboration in a way that would take into account not only the effectiveness of safeguards but also safeguards efficiency.*

*The lengthy and active cooperation between the IAEA and EURATOM facilitated the establishment in June 2011 of a working group, composed of representatives from the IAEA Department of Safeguards and the European Commission (EC), to review the existing cooperation arrangements between the IAEA and the EC and identify additional areas for enhanced cooperation.*

*In early March 2012, the working group prepared a draft report on the outcome of its work. The draft identified strategic areas for further cooperation which were expected to have the most impact on collaboration between the two organizations and recommended concrete actions that could be undertaken in the short term.*

*The IAEA and the EC noted that the draft is a good basis for the preparation of concrete proposals aimed at strengthening the cooperation between the IAEA and the EC to facilitate the implementation of the Safeguards Agreement (INFCIRC/193) and the NPA. The group was requested to continue to examine the possibilities for enhanced cooperation outside the current NPA arrangements but related to the implementation of safeguards in the NNWSs members of the EU.*

**Keywords:** SSAC; RSAC; IAEA, EC, EURATOM, cooperation; Reflection Group; NPA

## 1. Introduction

One of the key elements of the IAEA efforts to evolve safeguards implementation is the importance of cooperation between the IAEA and States to facilitate the implementation of safeguards. Under certain conditions, the IAEA may be able to rely on the support of State and regional authorities to a greater extent in the day-to-day implementation of safeguards while still maintaining its ability to draw independent safeguards conclusions. In order to do so, it is important that States and/or regions have competent authorities responsible for safeguards and enact legislation to enable the establishment and maintenance of effective systems of accounting for and control of nuclear material (SSACs/RSACs), including the capability to make independent verifications pursuant to such legislation. “It is in the interest of both the States and the IAEA to cooperate to facilitate the practical implementation of safeguards, and is explicitly required under CSAs. Effective cooperation depends upon States and the Secretariat sharing a common understanding of their respective rights and obligations”. [1]

The efforts by the IAEA are of particular interest in the European Union, where EURATOM acts as the Regional Authority for safeguards for the twenty five non-nuclear-weapon States (NNWSs) of the European Union (EU) as well as for the two European nuclear-weapons States (i.e. the United Kingdom and France).

Currently, EURATOM, pursuant to the EURATOM Treaty, and the IAEA, pursuant to the relevant Safeguards Agreements and Additional Protocols, apply safeguards to all twenty- seven EU Member States, all of which have Safeguards Agreements and Additional Protocols in force.

## 2. Enhancing IAEA/EURATOM Cooperation: The Early Years

EURATOM and the IAEA have a long history of cooperation which began with the implementation of the Safeguards Agreement (INFCIRC/193) in 1977. The Agreement provides that the IAEA, EURATOM and the NNWSs shall co-operate in order to facilitate the implementation of safeguards. It has a Protocol that amplifies certain provisions of the Agreement. The Protocol specifies the conditions and means according to which co-operation in the application of safeguards shall be implemented to avoid unnecessary duplication of the Community’s safeguards, effectively laying out the conditions under which EURATOM will operate as a Regional system of accounting for and control of nuclear material. An important element of the Protocol was the establishment of a Liaison Committee composed of IAEA and Community representatives whose tasks were to review safeguards implementation performance and examine the development of safeguards methods and techniques.

In 1992, after several years of experience with safeguards implementation pursuant to the Agreement and the Protocol, a “new partnership approach” (NPA) was proposed to further enhance the co-operation and co-ordination between the IAEA and the Community. The NPA Agreement was signed by Commissioner Cardoso e Cunha and the Director General Hans Blix in Brussels on 28 April 1992. [2]

The NPA brought more efficiency to the planning and implementation of inspection activities, while maintaining independence of both organisations and their ability to draw their own independent conclusions. The NPA promoted the use of commonly agreed safeguards approaches and inspection planning, procedures, activities, instruments, methods and techniques so as to avoid, as far as possible, any unnecessary duplication of efforts. The most important principle affirmed by the NPA and which forms the basis for the in-field inspection work was the principle of “one-job-one-person” whereby inspectors from each organisation work separately during inspections, but share inspection working papers and results, “supplemented by quality control measures to enable both organizations to satisfy their respective obligations to reach independent conclusions and required assurances”. [2, 3]

Cooperation was also extended to sharing analytical capabilities in order to reduce the number of samples taken, transported and analysed. Results were to be exchanged, and accepted, by each organisation.

The NPA also foresaw the common use of technology which could reduce the requirement for the physical presence of inspectors. For example, the mailing of closed circuit television (CCTV) tapes from facility surveillance systems to the IAEA was proposed at the time. This has evolved into the

digitally authenticated, remotely monitored and encrypted systems now common place in many nuclear facilities today.

The entry into force of Additional Protocols within the EU States and EURATOM, and the IAEA's ability to draw the broader conclusions regarding declared material and facilities as well as the absence of undeclared, provided the basis for the safeguards approaches for each State to be modified and in certain circumstances for the in-field inspection requirements to be reduced. This was achieved through comprehensive consultation between the IAEA and EURATOM over many months through the Liaison Committee, and ultimately resulted in the selective introduction of random and unannounced inspections at light water reactors, spent fuel storages, fuel fabrication plants, research reactors and enrichment plants throughout the EU. As a consequence of jointly implementing these arrangements, the IAEA has been able to reduce the number of routine inspections at these facility types throughout the EU.

## 3. Enhancing IAEA/EURATOM Cooperation: Current Considerations

The NPA was a significant step to better cooperation and coordination between the EC and the Agency which has led to increased efficiencies in the implementation of safeguards. However, after nearly twenty years of implementation, in July 2011 a working group consisting of representatives from both organizations (often referenced to as the Reflection Group) was formed and tasked with reviewing existing cooperation arrangements between the IAEA and the EU NNWSs to identify further possibilities for enhanced cooperation. One fundamental guideline for the group was that they should explore all possibilities within the existing legal framework, including the NPA arrangements, where cooperation could be enhanced.

In early March 2012, the group prepared a draft report on the outcome of its work. The report identified and elaborated on a number of underlying foundations that have to be in place for the IAEA and the EC to engage in enhanced cooperation, with the aim to make best use of the available competencies in each inspectorate. The report also contained a proposal for strategic areas which were expected to have the most impact on collaboration between the two organizations, and further identified concrete actions or tasks that could be undertaken in the near term.

### 3.1. Underlying foundations

The Safeguards Agreement (INFCIRC/193) already recognises the capability of the Community system of safeguards and urges the IAEA to take due account of it. For the IAEA to do so, it must be satisfied that safeguards activities are performed in an independent and competent manner.

The quality of safeguards conclusions is dependent on factors related to the competence of the organisation implementing it. This includes assurances that implementation processes and procedures are in place for all safeguards relevant activities, and that compliance is monitored, with any non-conformities being identified and addressed, with the aim of continually improving the performance of the safeguards system.

The organisation should ensure that individual staff members are free of any commercial, financial or other influences to their impartiality and judgement. They should have the appropriate levels of qualifications, training and experience to allow them to discharge their professional duties, and possess the ability to recognise any deviation or non-conformity which may occur in inspected facilities and to propose and implement corrective actions.

The results of inspection activities should be documented within an agreed timeframe and to an agreed format, such that each organisation can have confidence in the completeness and correctness of the information being exchanged.

One vehicle that allows an organisation to demonstrate these characteristics is a quality management system (QMS).

The decision to implement a QMS demonstrates an organisation's desire to document, monitor and continuously improve its processes and procedures in order to consistently produce a quality product.

The QMS can also help an organisation to demonstrate independence, impartiality and integrity of its staff.

### 3.2. Strategic Areas for Enhanced Cooperation

The Reflection Group identified and elaborated on the following strategic areas for enhanced cooperation in the implementation of safeguards in NNWSs of the EU, which are expected to have the most impact on collaboration between the two organisations:

- An International Standards Organisation (ISO) based QMS for EURATOM safeguards, including establishing synergies with the IAEA Safeguards Department's QMS;
- The more efficient management and use of common instruments, processes and procedures;
- The broader application of the principle "one-job-one-person";
- The exchange and utilisation of safeguards information.

### 3.3. The EC Quality Management System

In order for the European Commission to guarantee the quality of their safeguards activities and the conclusions that are drawn from these activities, it has already decided to implement a QMS. Both the IAEA Safeguards Department and EURATOM will have their own QMS [4]. However, parts of these QMS already have common elements, including joint procedures and processes, which facilitate common activities and allow the utilisation of each other's data and results.

### 3.4. The more efficient management and use of common instruments, processes and procedures

There is potential for more efficient ways to develop, manage and use common instruments, processes and procedures. Certain needs for increased efficiency in the area of common instruments have been identified before, such as the need for common acceptance criteria for safeguards instruments and the early involvement of the EC or of the IAEA in development activities undertaken by or for the other inspectorate. There remains however a number of areas in which further efficiency gains can be achieved.

One area identified to be explored with priority was the joint preparation, in-field use and subsequent verification of IAEA/EURATOM copper-brass common seals. The topic covers a well-defined, specific task currently performed jointly by the two inspectorates in the spirit of "one-job-one-person". The specificity of the topic could lend itself to an enhanced method of working in the near term. A review and optimisation of the scheduling of common seals verification activities in Luxembourg is being considered and could deliver resource savings if successfully implemented.

### 3.5. The broader application of the principle "one-job-one-person"

Initially, the "one-job-one-person" principle was implemented as part of the NPA arrangements of 1992, replacing the previous principle of "observation" and "Joint Teams". However, over time and in some cases the principle has become less practiced. One reason for this was a loss of knowledge of working arrangements due to replacement, rotation or retirement of experienced staff. New inspectors are often unfamiliar with the arrangements foreseen in the NPA, including the principle of "one-job-one-person".

Currently, efforts are being made to revive the very sound and efficient NPA principles to be used as standard practices within the EU, and also to explore new ways and areas for implementing the "one-job-one-person" principle to the advantage of both inspectorates. The expected advantages are:

- Enhanced efficiency and effectiveness of safeguards implementation;
- No duplication of activities;

- Less time spent in controlled areas (reduced dose uptake).

More potential gains may become possible from carrying the “one-job-one-person” principle to its maximum extent such as technical interventions or inspection activities that can be done by one organization alone supplemented by an effective QMS and appropriate quality control measures.

The success of the principle is dependent on the awareness by staff of both inspectorates of their responsibilities and obligations. Joint training plays a key role in this area.

### 3.6. The exchange and utilisation of safeguards information

All inspection activities carried out jointly or separately by the EC and/or the IAEA in any facility or LOF in the EU are documented, normally using agreed working papers. The papers are exchanged during every joint inspection.

For inspections where the IAEA is not present, the EC prepares an inspection report, often containing copies of all the relevant working papers, and sends a hard copy to the IAEA.

The current way of transmitting information in hard copy is not efficient and limits its use. In order to better utilise the data, ways to transmit information in electronic format or to access information stored in headquarters or on site, could be explored.

The IAEA could consider whether information from EC inspection reports, audit reports, letters to the inspected installations, material balance evaluations, could be useful as part of a wider range of information available about the nuclear activities in a State. Similarly information from IAEA activities could be of use to the EC.

## 4. Possibilities for Enhanced Cooperation in the Longer Term

The following topics were identified by the working group as possibilities for further cooperation in the long term, which were not included in the strategic areas:

- **Joint training** as a crucial element to improve the understanding the inspectorates have of each other’s systems, to ensure the smooth implementation of common safeguards activities and the coherent implementation of common safeguards approaches;
- **Channels of communication** to ensure that the cooperation between the inspectorates on planning and conducting inspections leads to real gains in efficiency;
- **Random inspections** and their implementation modalities to ensure that they can be implemented without creating undue burden or costs for the inspectorates yet still meeting the safeguards objectives;
- **Improved information exchange** between the two inspectorates to further increase the overall safeguards effectiveness;

## 5. Conclusion

The IAEA and EURATOM have been cooperating in safeguards implementation for forty years. This cooperation has developed since 1977 from the IAEA observing the inspections of EURATOM, through joint inspections at nuclear facilities and LOFs utilising the “one-job-one-person” principle, to more advanced safeguards approaches utilising random inspections, unannounced inspections and complementary access.

The enhanced cooperation between the IAEA and EURATOM could lead to mutual benefits for both organizations and their Member States by allowing the IAEA activities being increasingly focused and efficient, making safeguards implementation more cost effective and minimizing its impact on nuclear operations without compromising the ability to draw independent conclusions. A recently established working group reviewed this cooperation and explored the conditions under which the IAEA could

"take due account of the effectiveness of an SSAC/RSAC system of safeguards" by identifying underlying foundations which allow the IAEA to validate SSAC and RSAC provided information. The working group has identified specific areas where enhanced cooperation would be possible and beneficial and has suggested areas where more could be done within the existing legal framework.

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# Nuclear Safeguards and Non Proliferation Education and Training, initiatives by ESARDA, INMM and JRC

**W. Janssens (1), M. Scholz (2), T. Jonter (3), M. Marin Ferrer (1), A. De Luca (1)**

(1)Institute for the Transuranium Elements
Joint Research Centre, European Commission
Via Fermi, Ispra 21020 (VA) Italy
E-mail: willem.janssens@ec.europa.eu
(2) US National Nuclear Security Administration (NNSA)
(3) Stockholm University, Sweden

***Abstract:***

*The subject paper reviews and discusses some aspects of education and training in the field of nuclear safeguards and non-proliferation. This is largely inspired on the INMM ESARDA Working Group meeting in Aix-en-Provence in November 2011 and includes additional elements esp. w.r.t. outreach activities (i.e. capacity building efforts outside Europe). The paper will also refer to the NUSASET web-site, its potential and use. In addition, reference is made to the INMM Student Chapters in mechanism, which is a specific initiative to foster students to work and study in the area of Nuclear Materials Management*
*In Aix-en-Provence a number of challenges and requirements in the subject field were identified :*
- *Standardisation e.g. min. educational and training requirements*
- *Analysis of the stakeholders and their needs e.g. training passports*
- *Infrastructure, tools and approaches used for education and training*
- *Knowledge management and mobility of information and students*
- *Cross fertilization with other fields : export control & security*

*In the mean-time, extra emphasis has been put on capacity building, including education outside Europe. Both for new-comers in the nuclear field and for experienced services, the nuclear safeguards and non-proliferation remains an area with typically little visibility and/or awareness. Dedicated efforts are thus ongoing to promote the education and training in this area e.g. in South-East Asia and elsewhere. Such capacity building efforts are supported under the Instrument for Stability, i.e. financed by DG DEVCO of the European Commission, and will in the future be enhanced further under the so-c alled CBRN Centres of Excellence initiative, in which e.g. the European Commission JR C plays a key role (e.g. in th e areas of RN security and export control)*
*The experience gained in the regular ESARDA course on nuclear safeguards and non-proliferation, held annually at JRC-Ispra and hosted by the Nuclear Security Unit, and its "export" outside Europe, held in February 2013 in Malaysia, will also be reported upon..*

**Keywords:** education; training; nuclear safeguards; ESARDA; INMM

## 1. Introduction

Training and Education are key activities to stimulate a new generation, underpin capacity building, maintain competences and allow proper implementation of nuclear safeguards, non-proliferation and nuclear security. The need for dedicated efforts in this field was recognized, also internationally, ten years ago [1,2], in parallel to the dwindling knowledge in the nuclear field in general, due to both the lack of new students and the lack of professionals choosing e.g. at mid-career to move into nuclear in general and nuclear safeguards, non-proliferation and nuclear security in particular. [3] As a consequence, when colleagues with extensive experience leave the field, it is very unlikely not only to find similar competence on the job-market but even to recruit a new generation of colleagues who know about nuclear safeguards, non-proliferation and nuclear security [4,5]

Similarly it is a significant challenge for newcomers to the civil nuclear fuel cycle to identify in an efficient way all requirements, competences and skills needed for a safe and secure operation of one or more fuel cycle facilities. This has become a clear challenge, mainly outside the EU and US, and some dedicated outreach programmes, including those dealing with universities and competence/training centres have been set-up but much more needs to be done to keep the momentum and assure at least a certain level of sustainability.

Thus important efforts are needed to both invest in in-house training (e.g. at IAEA, US-DoE and the European Commission) for this purpose and to further motivate university professors and training centres to include the topic of nuclear safeguards, non-proliferation and nuclear security in their curricula. The challenge is all the more difficult, due to the multidisciplinary nature of the field of work, going from basic physics, radiochemistry, nuclear measurements, applied statistics and advanced information technologies, all the way to international political studies and legal instruments.

Whereas ESARDA organizes e.g. yearly a dedicated course on Nuclear Safeguards and Non-proliferation (in 2013 even twice), INMM has a variety of activities for students and universities, e.g. the fostering and support of building Student Chapters of like-minded people feeling that discussing/analyzing the science together can only bring benefits to both. Likewise, the US-DoE/NNSA Next Generation Safeguards Initiative has included a robust Human Capital Development Program which has educated and trained over 700 students in non-proliferation and safeguards in five years, including a number of foreign nationals who have partaken in NGSI university courses, short courses, and fellowship opportunities. The program has also funded INMM initiatives including the JD Williams Student Paper award and hundreds of INMM and INMM-ESARDA meeting participants, including over 150 students

## 2. INMM-ESARDA initiatives.

The last two INMM-ESARDA meetings held respectively in 2008 in Japan and in 2011 in France have both included a special topical session on Education and Training. These sessions were typically very well attended by a variety of stakeholders such as inspection bodies (Euratom Luxembourg and IAEA), national authorities, professional organizations, some industry representatives and especially colleagues which are active in teaching either in professional schools or at universities.
In line with the priority given to training and education, both by INMM and ESARDA a number of actions came out of these workshops, which led, amongst others, to the set-up of a dedicated international group, baptized NuSaSET : Nuclear Safeguards and Security Education and Training (www.nusaset.org)
In Aix-en-Provence, the Education and Training working group reviewed the actions from 2008, discussed the results achieved, analysed the remaining and new challenges, opened a number of new avenues and thus came out with a new action list, which was presented to the plenary meeting at the end of the meeting.

### 2.1. Achievements since 2008.

Since the last WG on Training and Education, held during the INMM-ESARDA meeting in Tokyo in 2008 [6], and during which a substantial action list was agreed upon, a lot of progress has been made. This was discussed during the WG in Aix-en-Provence 2011, mainly during the first afternoon, and the following main points are retained :

- In line with the ESARDA WG on Training and Knowledge Management, and in addition to the well-functioning student chapters at INMM, there was the formation of a dedicated INMM Education and Training Committee at the end of 2010
- Through the outreach activities of pro-active INMM and ESARDA members, there was in recent years a good initial ANS and ENS engagement on nonproliferation and safeguards issues
- As a full novelty since 2008, it is very valuable to be able to report that five export control trainings for the IAEA Department of Safeguards have been organized in the last years, confirming the notable signs of cultural change also in the safeguards approaches at IAEA
- Also in the IAEA nuclear security department substantial progress in educational activities has been made and a very active programme is being run

- When looking on a world-scale it is gratifying to note that there is an increasing number of university programs and courses in the area of safeguards and nonproliferation
- Several new nuclear training centers came into operation, typically focusing on nuclear security but including also modules on safeguards, non-proliferation and nuclear security
- In addition, it is clear from the reporting of the different partners, that there is a growing recognition of the importance of nuclear security and nonproliferation challenges by emerging nuclear countries
- Some progress has been made w.r.t. quantifying and qualifying needs of T&E but further work is deemed needed
- For sharing information, e.g. on T&E opportunities, it is noted that there is an increasing use of social media and technology
- W.r.t. internship opportunities, a three-fold increase in number of safeguards internships in the United States was achieved since 2008 whereas there is a stable but limited number in Europe
- The creation of new professional development networks (e.g. NuSaSET, INENS, NGSPN) for the next generation in nuclear security and safeguards was welcomed and was discussed in some detailed presentation in the 2011 WG meeting.

## 2.2. Contributions to WG4 of the INMM-ESARDA 2011 Aix-en-Provence meeting.

Because of the multidisciplinary nature of the topics of safeguards, non-proliferation and nuclear security, it is quite a challenge to run a WG during the INMM-ESARDA meetings with specialists only dedicated to training and education. In fact, many colleagues who attended also other WG's were contributing actively to WG4, also when not being present all the time. We also had topical contributions from colleagues connected per videoconference. We got a very good participation from all the different stakeholders : universities, professional training centres, international student networks, research and development, industry, operators and safeguards authorities / inspectors.

The "new" questions which had been put forward for this WG 4 meeting were phrased as follows :
- Do we fill the gap in the current nuclear safeguards and security E&T?
- How do we prepare best for the future nuclear safeguards and security E&T esp. in emerging nuclear countries?
- Which best practices, for E&T concepts, can be shared and transferred and how do they differ between academic and professional teaching? Discuss the different approaches with pro and cons and derive best practices from experience, student feedback and industry needs. (e.g. Train the trainer, In-field training, Class-room training, Internships, On the job learning, Retraining / long-live learning etc)
- Which efficient E&T tools are used for which purpose, at different stages in the learning cycle and which ones remain to be developed? (e.g. E-Learning, Virtual reality, Hands-on Exercises, Theoretical case-studies, Role-plays etc)
- How can we address the multi-facetted challenge on Knowledge Management in view of E&T in nuclear safeguards and security? (e.g. Manage the knowledge of staff and resources, Maintain historical knowledge, Information gathering, structuring, sharing, use , Collaborative environment (e.g. wiki's, forum, etc))
- Which synergies can we valorize between safeguards and security E&T (also looking at safety)?

The following speakers contributed to the programme
- M. Scholz : US-DoE, US
- W. Janssens : JRC-ITU, Italy
- S. Zero : AREVA, France
- M. Goettsche : Hamburg University, Germany
- S. Grape : Uppsala University, Sweden
- R. Berndt : JRC-ITU, Italy
- S. Synetos: DG ENER Euratom Luxembourg
- E. Martika : STUK, Finland
- T. Jonter : Chair ESARDA WG TKM – Stockholm University, Sweden
- J. Baute : IAEA : Director SGIM, Vienna
- F. Sevini : for the Deputy chair ESARDA WG TKM – JRC, Italy
- Mr. Damamme : Institut Etudes Superieures Nucleaire de Defence, France
- C. Guet : I2EN : International Institute of Nuclear Energy, France
- Pete Heine : ANL : Export Control Training, US
- Jong UK Lee KINAC: Intl. Training centre – S. Korea

- L.V.Bril :EU- External Action Service, Brussels, Belgium
- Meena Singelee: INENS, UK
- Craig Everton : Asian-Pacific Safeguards Network , Australia
- A. Braunegger : IAEA : Nuclear Security Department

## 2.3. Actions defined in the WG4 of INMM-ESARDA 2011.

1. Establish INMM-ESARDA minimum standard for safeguards education and training modules
    a. Develop a recommendation document based on lessons learned and consensus of key stakeholders
    b. This can feed innovative approaches into education and training cycle
    c. This is expected to benefit both emerging nuclear countries, as well as promoting cultural change
2. INMM-ESARDA should examine ways to promote making safeguards/ nonproliferation a mandatory element of nuclear engineering curricula
    a. Raise awareness in academic circles
    b. Proactive outreach to university faculty
    c. Tap into student INMM chapters
    d. Share materials of success cases
    e. Develop alongside security and safety curricula/ models
3. INMM-ESARDA should foster the availability of funding for E&T activities
    a. Close interaction with U.S. DOE/NNSA/NGSI and EC, which have these funding capabilities
    b. Reach out to additional national and regional sponsors
    c. Relevant for both domestic and international applications
4. Fostering exchange of students and trainees (geographically, cross-disciplinary)
    a. Tapping into existing national internships, courses, fellowships
    b. Provide access to sensitive facilities under international safeguards
5. Essential to guarantee access to relevant nuclear infrastructure for training purposes
    a. Dedicated training facilities
    b. Access to operational plants
    c. In addition, need to further develop simulation, virtual reality tools, and e-learning modules
6. Expand INMM-ESARDA interactions with other networks and stakeholders
    a. APSN, ABACC, NGSI, nuclear security training centers
7. Knowledge Management
    a. Not enough emphasis on “tools for the brain”
    b. Option of a NuSaSET portal
    c. Foster exchange and dissemination of course materials
    d. Build a more collaborative approach to support cultural change
8. Stimulating further cross-fertilization between export controls and safeguards
    a. Content-wise
    b. Training methodology (multiple layers of engagement)
    c. Risk-based prioritization
    d. Outreach to promote compliance
    e. Front-line inspector vs. investigator role
9. Promote training geared towards teamwork, making better use of different experiences in a group dynamic; collaborative analysis
10. Address special focus groups, including diplomats, media, or interpreters
11. Deepen integration with NGOs, including proactive engagement to build partnerships in E&T

## 3. NUSASET initiative: A bottom-up approach for more effective knowledge sharing

In the past decade, the problem of transferring the knowledge of nuclear professionals to the next generation has become a serious issue that needs to be solved effectively, in the shortest possible time.

This new generation of nuclear professionals does not seem to be enough to replace the past work force, and what is more concerning is the fact that passing on the life-long expertise to them is even more of a challenge, as was clearly stated during IAEA's 2004 conference [1] on 'Managing Nuclear Knowledge: Strategies and Human Resource Development' where it was stated that: "Many experts around the world are retiring, taking with them a great deal of knowledge and corporate memory. The people retiring are those who can answer questions easily and have tacit knowledge that was not extracted from them previously."

With this in mind, NuSaSET has been following a growing trend where education and knowledge transfer is a rapidly shifting paradigm, from a more traditional top-down approach to an open collaborative approach, thus breaking away from the conventional ways that have been in place for more than 50 years in such highly technical and specialised fields. By this, we are not intending to state that Web 2.0 is the perfect substitute. Nevertheless, common learning approaches have revealed a lack of interaction and communication between the participants, which has lead to the approach of technology-enhanced learning (TEL) [7]. What is being said is that this concept puts the learning process up front and considers technologies as supporting means, helping to accelerate information flows and reduce barriers.

In fact, digital social networks and user-generated content are the driving force of today's participative web [8]. The basis of this approach is the user participation and the ease of distribution of user-generated content. As such, NuSaSET has adopted Web 2.0 technologies for effectively sharing information on the Web where users can actively create, store, edit, access, share and distribute the content to larger audiences at all times.

Specifically, NuSaSET has set-up a dedicated Social Network to complement the existing website. The Social Network, based on a Joomla Platform, provides the means to create one's profile, by choosing between a Student and Professional profile. In this virtual "agora" users can meet and interact without the conventional social boundaries, easing interaction amongst the community. For example, if a student or a researcher nowadays needs some information on a specific topic, or would like a second opinion on a topic he may not be familiar with, he can simply refer to his online community, where he is likely to receive answers from his peers in a short time span.

On top of the benefits that social-network technology provides, NuSaSET is in the process of developing the website as a reference point for its users. Amongst these are the document repository and online library, containing all the most important treaties, interesting topical articles and scientific papers that can assist any professional with work research.

Having mentioned all the advantages of adopting such a bottom-up approach, NuSaSET and the ESARDA TKM Working Group are well aware that this approach does have some weaknesses. For example the bottom-up approach, complemented with technology, relies highly on the willingness of employees to share and to contribute, which at times seems easier than it actually is to implement. The human factor is the biggest hurdle to overcome. NuSaSET must first be able to convince its audience of the strategy adopted and of the benefits it can provide to both online communities (students and professionals). We do, however, feel that some existing initiatives put in place by the TKM WG, such as the ESARDA Course, can be leveraged effectively to recruit new members to contribute to this initiative. In fact, to this date NuSaSET can rely on a "niche" community of nearly 300 registered users, who are using to their advantage the services being offered by the portal.

## 4. ESARDA Courses in 2013

The absolute flagship activity of the ESARDA WG TKM is the annual Nuclear Safeguards and non-proliferation course, held typically at the JRC premises at JRC-ITU Ispra and managed by the Nuclear Security Unit. In total 11 sessions of this course have been given so far (of which 2 not taking place in Ispra) and each time the course is significantly oversubscribed and interested participants have to be refused (to stay at maximum handable number of participants per session i.e. 60).

In February 2013 the course has been held for the first time outside Europe, based on funding received through the DG DEVCO ran Instrument for Stability. This allowed JRC (under contract with DG DEVCO) to set-up a full week session in Kuala Lumpur, Malaysia. Over 30 students from 7 South-East Asian countries participated very actively, whereas teachers were not only coming from the JRC but from EU (some of those teaching regularly in Ispra), from Australia, Japan, S. Korea, US and IAEA.

The ESARDA courses receives typically are very positive feedback, a number of participants expliclily regretting that it cannot be longer. The format of the course, where every day there is a visit to one of the safeguards laboratories on the Ispra site, is highly appreciated also.

The course syllabus, which was compiled about 5 years ago and of which more than 2000 printed copies were distributed so far, is based on contributions from many ESARDA working groups. It could be revised in the near future, to incluede more of the newest thinking w.r.t. internatinonal safeguards, such as the state evaluation concept and possibly new chapters such as disarmament verification or multilateral fuel cycle facilities and concepts. Because of the very dense programme already in the one week on the other hand, not many new topics can be taken up, so choices will have to be made.

The ESARDA course continues to be recognised with 3 ECTS (European Credit Transfer System) points which thus allows university students, Europe wide, to apply for this course and integrate it in the curriculum, when also passing the exam and making the essay.

## 5. INMM Student Chapter opportunities

In 2005, the INMM created one of the first of many university student INMM Chapters at Texas A&M University. This first endeavour in engaging university students in the area of responsible nuclear materials management (i.e., nuclear material safeguards, nuclear security, and nuclear non-proliferation) laid the ground work for the 13 current student INMM chapters around the world. These chapters, ranging from policy universities (such as at the Monterey Institute for International Studies) to universities with nuclear engineering departments (e.g., University of Tennessee – Knoxville) and to universities outside the United States (e.g., the Jordan University of Science and Technology), provide opportunities for undergraduate and graduate-level students to interact with other students that hold similar interests for safe and secure nuclear materials management. Events have been coordinated by each student INMM chapter throughout the respective academic year to encourage lively discussions and deeper education in needs relevant to the nuclear power and non-proliferation industries. Together, these chapters have created a global network of student and younger professionals interested in furthering the nuclear non-proliferation and safeguards cause.

Specifically, student INMM chapters have worked closely with the INMM technical divisions in hosting workshops at their universities in order to expose other students to the current state-of-the-art and to assist the INMM's engagement of university-level students and faculty in these areas. Previously, student INMM chapters have joined with the Non-proliferation and Arms Control Technical Division and the International Safeguards Technical Division to co-host workshops and seminars apart from the INMM Annual Meeting held every July in the United States.

An important aspect of the student INMM chapters has been the involvement of appropriate faculty whom serve as a mentor for those student INMM members, without which, the sustainability outlook of the student INMM chapters is bleak. Faculty engagement is essential for the continued success of interacting with university students in this area. It can be understood that students conduct research and deepen their understanding of safeguards and non-proliferation via professors with strong inroads to the industry and that professors serve as the primary link with the INMM and other professional

networks (e.g., ESARDA). Furthermore, when university faculty host professional visitors from laboratories or other international research institutes, this provides the opportunities for that student INMM chapter to hold a meeting where the professional visitor is able to give a presentation on their work, organization, or the INMM specifically.

Though the activities of the various student INMM chapters do not, in particular, meet the specific actions conveyed in Section 2.3, there is no reason that they could not. It has been discussed with various student INMM chapter presidents and faculty advisors to begin addressing those actions but nothing has yet been institutionalized. However, the student INMM chapters have the ability to meet some actions better than others. For example, just as the student American Nuclear Society chapters in the United States have led to close collaborations between the nuclear industry and the respective university, so can the student INMM chapters with similar (if not the same) nuclear industry. Additionally, with regards to the larger international organization, the American Nuclear Society has taken the initiative in preserving nuclear engineering knowledge via publishing textbooks and other educational material whereas, the INMM has not but very well could serve this role for the purpose of knowledge management.

In all, the student INMM chapters (and the INMM as a whole) should and can do much more to meet the needs of properly educating the next generation of safeguards and non-proliferation subject matter experts. The foundation has already been established with a number of competent and successful student INMM chapters and it is wise to establish more around the world where there is interest and capability. The involved students could benefit from such a vast global network in many ways (expanding their knowledge of the topics and beginning their international networks of contacts) but the industry itself would benefit just as much.

## 6. Update on US-fostered E&T activities

As discussed, the challenges facing the safeguards and nonproliferation community have not been limited to a particular country or region; rather, a convergence of factors has been challenging the entire international nonprolfieration community. At the IAEA, the combination of growing workload, increasing complexity of the work, and anticipated retirements pose particular challenges to the IAEA Department of Safeguards. In 2008, IAEA officials estimated that more than half of IAEA senior staff would retire by 2013. In the same year, the Agency also reported that less than 20% of the IAEA's safeguards inspectors were under 40 years of age [9]. Similarly, a U.S. safeguards workforce study funded by the Next Generation Safeguards Initiative (NGSI) in 2009 found that more than 80 percent of the international safeguards experts at U.S. National Labs would be leaving the workforce within 25 years [10].

DoE-NNSA's NGSI Human Capital Development (HCD) program has, since its inception in 2008, sought to address these gaps and identify and train a new cadre of safeguards experts. The NGSI HCD efforts have endeavoured to build a continuous pipeline of new talent by recruiting, educating, training, and retaining the next generation of talent. The main components of this pipeline include university engagement, curriculum development, laboratory internships, short courses similar to the ESARDA course, graduate and post-graduate fellowships, professional development, and INMM/ professional conference support.

In the context of the goals of the E&T working group, it can be noted that while NGSI has a domestic focus, its underlying purpose is international. Particularly since co-sponsoring a Human Capital Development workshop with JRC-Ispra in 2009, additional opportunities to collaborate and contribute to international efforts has become more of an emphasis. Laboratory internships, six of its eight short courses, and post-graduate lab positions are open to foreign nationals. In 2013, NGSI is funding one of its Nuclear Nonproliferation and International Safeguards (NNIS) graduate fellows to complete his practicum at JRC-Ispra. In the early summer of 2013, NGSI will support a young professional to complete work at Forschungszentrum Juelich in North Rhine-Westphalia, Germany.

Beyond fostering these types of professional exchanges, the U.S. is also addressing E&T working group recommendations by: including export control as a topic in safeguards curriculum development; exploring ways to post training modules online; completion of a safeguards textbook for use in both policy and technical coursework; connecting with the International Nuclear Security Education Network

(INSEN); and development of facility training opportunities. It is hoped that the U.S. experience and efforts toward nonproliferation and safeguards E&T domestically and in partnership with other international stakeholders can continue to blossom to best support the venerable goals of the INMM-ESARDA working group.

## 7. Funding and outreach activities

As already shortly referred to in the previous paragraphs a variety of funding opportunities exist for countries, partners and stakeholders who want to invest in nuclear safeguards and non-proliferation education and training. This can be realised in a number of ways.

Funding for universities to develop a masters programme (or include safeguards/non-proliferation courses in existing curricula). This is e.g. co-funded at the Chualalongkorn University in Bangkok, Thailand, by the Centres of Excellence Initiative of the European Commission (for both Thai and regional students)

Funding for specific E&T initiatives, like the aforementioned ESARDA course on nuclear safeguards and non-proliferation (in Kuala Lumpur, but inviting all South-East Asia countries to attend) which was paid by the Instrument for Stability of the European Commission

From the US, the NGSI, as described above, is also a first opportunity by excellence providing financial support for (overseas) grants and educational opportunities.

In fact, these examples above illustrate that Education and Training are a central place in the current financial instrument priorities. The concept of “CBRN Centres of Excellence” has e.g. as key objectives to establish good quality, in-house and sustainable capacities in the CBRN area, with a major emphasis on education and training (i.e. as investment for the future)

Important are also the in-kind contributions as e.g. by the nuclear security at JRC-ITU which is assuring the organisation for the annual ESARDA course and provides regularly grants to doctoral and post-doc students to work in the area of nuclear safeguards and non-proliferation, which allows the students not only to gain knowledge but also to perform in a real work-environment

## 8. Conclusions

Five years after the meeting in Japan and two years since the meeting in France, it is important to take a look at the progress made in the intervening years, where we can be doing better, and where we may want to edit or improve upon previous recommendations. As underscored by section 3 of this paper, a major success towards these identified goals has been found in NuSaSet, particularly its web portal.

As a next step, it is the recommendation of these authors that each item previously identified by the INMM-ESARDA working groups (see section 2.3) be assigned an action officer with responsibility for promoting this goal and to develop practical, concrete recommendations for its implementation. Members of the NuSaSet network—including representatives of INMM, ESARDA, NGSI, universities, and other training centers -- are well poised to take on such a leadership role to bring more prominence to individual international E&T goals. This will allow future meetings to build upon the work done in the mean-time.

Both the ESARDA WG TKM and the INMM Education and Training Committee and Student Chapters are very valuable mechanism to further promote these initiatives and the active participation, world-wide, of university professors and academic “champions” is being sought for.

## 9. Acknowledgements

The authors would like to acknowledge Veronique Berthou, JRC-Ispra, Italy and Claudio Gariazzo, Texas A&M University, US for their valuable contributions to this paper and the very efficient collaboration to the realisation of the ESARDA and INMM education and training goals.

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# Enhancing the Capabilities of the IAEA Safeguards Analytical Services: New laboratories and techniques to meet future analytical challenges

Gabriele Voigt, Mark Scheland

International Atomic Energy Agency
Office of Safeguards Analytical Services
Wagramer Strasse 5, P.O. Box 100
1400 Vienna, Austria
E-mail: g.voigt@iaea.org, m.scheland@iaea.org

***Abstract:***

*The International Atomic Energy Agency (IAEA) is conducting the project Enhancing Capabilities of the Safeguards Analytical Services, ECAS, with the aim to modernize, expand and, where necessary, replace laboratory facilities and supporting infrastructure in Seibersdorf, Austria. At these laboratories, samples of uranium, plutonium, spent fuel dissolver solution and high activity liquid waste materials, as well as environmental swipe samples collected by IAEA inspectors, are processed and measured in order to determine the elemental and isotopic composition of the radionuclides they contain.*

*In parallel with the laboratory construction project, an expansion of the IAEA's Network of Analytical Laboratories, located across a number of Member States, is progressing well.*

*The poster supported by this report describes the IAEA Safeguards Analytical Laboratories' principle tasks, methods and capabilities, planned new features, and the support being provided by the European Union and other IAEA Member States to the ECAS project.*

***Keywords:*** analysis; sampling; laboratory; IAEA

## 1. Introduction

The International Atomic Energy Agency (IAEA) is responsible for deterring the proliferation of nuclear weapons by detecting early the diversion of nuclear material or technology and by providing credible assurances that States are honouring their safeguards obligations. The analysis of nuclear material samples and environmental samples collected during IAEA in-field verification activities is an essential component of safeguards implementation. Samples collected by IAEA inspectors are analysed at the IAEA's Safeguards Analytical Laboratories, composed of the **Nuclear Material Laboratory (NML)** and **Environmental Sample Laboratory (ESL)**, both located at Seibersdorf in Austria, and in a network of laboratories in Member States. A third organisational unit of the IAEA, the **On Site Laboratory (OSL)** at the Rokkasho Reprocessing Plant in Japan, is a joint facility staffed by the IAEA and host nation scientists who conduct analysis of nuclear material samples from the reprocessing plant.

## 2. Safeguards Sample Taking and Analysis

In conducting in-field verification activities, IAEA inspectors request nuclear facility operators to collect and provide *nuclear material samples* from various points of the nuclear fuel cycle. Inspectors may also collect *environmental samples* by swiping surfaces at various locations.

The nuclear material samples, received mostly in solid or liquid form, are subject to sophisticated analysis. Scientists in the NML or ESL, unaware of the country in which samples were obtained, focus on the detection of uranium and plutonium and determination of their concentrations and isotopic

compositions. The nuclear material analyses provide a powerful tool for supporting conclusions as to the correctness of each State's nuclear material declaration. The analysis of nuclear material particles detected in environmental samples can provide indications as to the completeness of a State's declaration. Together, these analyses allow the IAEA to draw conclusions on whether a State is complying with its safeguards obligations.

In carrying out this work, the IAEA Safeguards Analytical Laboratories participate in and coordinate the global Network of Analytical Laboratories (NWAL), which includes an additional twenty-one laboratories located in nine different IAEA Member States as well as the European Union. The ESL receives and screens all swipe samples but shares that analytical workload with its NWAL partners. Apart from the IAEA laboratories in Austria, as of May 2013, seven institutions in five EU countries and the European Commission were actively supporting the analytical work.[1]

## 3. Nuclear Material Laboratory

As part of the verification of a State's declarations, the NML (in continuous operation since 1976) receives samples consisting of uranium, plutonium, diluted samples of spent fuel dissolver solution and high activity liquid waste materials from points along the nuclear fuel cycle, which it then processes and measures. Analytical chemistry, radiometric techniques and mass spectrometric techniques are used to determine the elemental and isotopic composition of radionuclides found in the samples.

Strict quality control is essential for maintaining confidence in the results. This is attained through the use of certified reference materials and analytical methods compliant with internationally established quality assurance practices, as well as through the IAEA's participation in numerous inter-laboratory comparison programmes. Confidentiality is strictly maintained; the laboratory receives each sample in an anonymous, bar-coded container accompanied by a set of analytical requests from the evaluators in the IAEA's Department of Safeguards.

## 4. Environmental Sample Laboratory

The IAEA began its environmental sampling programme in the mid-1990s. The ESL is an extensive clean facility with functions that include the preparation of environmental swipe kits for sample collection, sample pre-screening, chemical processing and analysis of samples. Environmental samples can be any material which might contain trace evidence of nuclear activities, for example biota or soil, but most commonly, samples are in the form of dust collected on specially prepared cotton swipes.

Mass spectrometers are used to determine the isotopic composition of uranium or plutonium contained in samples in the nanogram to femtogram range. The Large Geometry Secondary Ion Mass Spectrometer (LG-SIMS), brought into service in 2011, provides a powerful analytical tool for the isotopic 'fingerprinting' of individual uranium particles. Expanded analytical capabilities for refined plutonium and other minor actinides detection are planned for the future. The IAEA's first Multi-Collector Inductively Coupled Plasma Mass Spectrometer (MC-ICP-MS) employs a system designed specifically for high quality and high sensitivity detection of all of the isotopes of either uranium or plutonium in bulk samples.

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[1] Institute for Transuranium Elements (EU JRC ITU)

Institute for Reference Materials and Measurements (EU JRC IRMM)

Czech Republic - ÚJV Řež, a. s. (REZ)

France - Commissariate a l'Energie Atomique (CEA)

Germany - Forschungszentrum Jülich

Hungary - Institute of Isotopes of the Hungarian Academy of Sciences (MTA/IOI)

United Kingdom - National Nuclear Laboratories (AWE)

Three further European laboratories participated in proficiency qualification testing for nuclear material analysis in spring 2013.

## 5. Inspection, Analysis and Evaluation in Close Partnership

Laboratory staff members provide comprehensive training for IAEA safeguards inspectors, for example, in the procedures for collecting environmental swipe samples and mitigating cross-contamination, or the sampling of nuclear material items in order to achieve representative samples.

The Safeguards Analytical Laboratories keep pace with technological developments through the use of state of the art instruments, frequent consultation with other experts in the field, and the support of relevant Member State support programmes. The laboratories' mechanical and electronic workshops, equipped with advanced machinery, customize laboratory hardware to meet the specific requirements of sampling conducted during inspections and safeguards analysis.

## 6. Modernizing Laboratory Facilities, Improving Security

The IAEA is conducting the project *Enhancing Capabilities of the Safeguards Analytical Services*, ECAS, with the aim to modernize, expand and, where necessary, replace laboratory facilities and the supporting infrastructure in Seibersdorf. The new Nuclear Material Laboratory (NML) building, nearing completion thanks to significant financial support from individual EU States and the EC among other donors, will provide a secure, flexible, fit-for-purpose facility in which to meet its share of the IAEA's analytical workload. Funding through the EC specifically is supporting the establishment of the NML's laboratory spaces for plutonium, "hot" environmental swipe samples, calibration and other "low-level" activities, trace and impurity characterization, and thermal ionization mass spectrometry (TIMS), as well as the sample handling logistics area. Eight EU States have further supported the ECAS project with national contributions as well.[2]

## 7. Partnership with the European Union

EU-IAEA cooperation in research and development (R&D) and EU material contributions to the Agency's safeguards analytical services are very significant. EU Member State experts and scientists in the EU Joint Research Centre (JRC) have been consulted in the design of the NML building and the site's physical protection concept, and the EC Directorate General for Development has contributed €10 million over two years to sponsor the construction and outfitting of specialized areas within the NML, as noted above.

Examples of EU-IAEA R&D cooperation specific to safeguards implementation include:

- The JRC's Institute of Transuranium Elements (ITU) is a key laboratory of the NWAL for nuclear material and environmental swipe sample analysis, and has recently brought into service a LG-SIMS for this purpose. ITU has developed an innovative in-field destructive analysis method and made it available to IAEA inspectors to perform in-situ analysis of concentration and enrichment of U fuel pellets.

- The JRC's Institute for Reference Materials and Measurements (IRMM) takes an active part in the ongoing quality control programme for the safeguards analytical services. These activities are important contributions to the overall NWAL quality control and ensure a high level of analytical performance in terms of accuracy, precision and dependability for the evaluation of safeguards samples.

- IRMM developed a superior technique to measure isotope ratios of uranium samples, which in combination with a new generation of certified reference materials is now in use at the IAEA's Nuclear Material Laboratory.

[2] The contributors are Czech Republic, France, Germany, Greece, Ireland, Netherlands, Spain and the United Kingdom. National contributions have ranged from €5000 to over €5 000 000.

## 8. Conclusion

Anticipating a continuing growth in the amount of nuclear material under IAEA safeguards worldwide and innovations in analytical techniques and instruments, the IAEA is constructing and bringing into service new laboratory facilities. The buildings are internally adaptable and conceived, in conjunction with their supporting site infrastructure, to meet international standards and guidelines for personal safety and the physical protection of nuclear material. IAEA Member State expertise, including prominent contributions from Europe, has been essential in the conceptualization and equipping of these facilities and in the validation of analytical results. Intensive scientific partnerships with European institutions will characterize the work of the IAEA Safeguards Analytical Services in the new laboratories as it has over more than five decades.

## 9. Acknowledgements

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# IAEA Safeguards Quality Management System

**Darko Korošec, Richard McCullough, Haroldo Barroso Jr., William T. Stanley, Erika Du Toit, Kenneth Y. Aoki, Debra Weisweiller**

International Atomic Energy Agency
Department of Safeguards
Wagramerstrasse 5, 1400 Vienna, Austria
E-mail: d.korosec@iaea.org

***Abstract:***

*The International Atomic Energy Agency (IAEA) Department of Safeguards quality management system (QMS) supports the Agency's commitment to providing soundly-based safeguards conclusions regarding the peaceful use of nuclear material. The focus of the QMS is to enhance the effectiveness and efficiency of safeguards implementation through a process-based approach, having designated roles and responsibilities for the system and establishing and maintaining quality management tools to implement the system. The QMS plays a special role ensuring consistency within the Department with the use of approved policies and procedures that are available to all staff. This paper describes how the various elements of the Department's QMS support safeguards implementation.*

**Keywords**: quality management system, safeguards processes, condition report, internal quality audit, document control.

## 1. Introduction

The International Atomic Energy Agency (IAEA) is committed to providing soundly-based safeguards conclusions to the Board of Governors and the international community regarding the peaceful use of nuclear material. To support this commitment, the IAEA Department of Safeguards took the strategic decision, over a decade ago, to establish and implement a quality management system (QMS) that conforms to the requirements of the ISO (the International Organization for Standardization) 9001 standard.

A quality policy statement, signed by the Deputy Director General, Head of the Department of Safeguards (DDG-SG), was prepared and distributed to all staff to explain the main objectives of the QMS and clarify responsibilities with its implementation and maintenance. The policy statement is displayed in poster format throughout the Department as a visible reminder to staff of the DDG-SG's commitment to the QMS.

In accordance with QMS principles, the high-level business processes representing the Department's activities were defined. The processes were categorized into three types: core, management and support. These processes form the basis of how the Department operates and show how a range of inputs contribute to the drawing of soundly-based safeguards conclusions and reports on States' compliance with their safeguards obligations, thus meeting Member State expectations. The interaction between the process types enables the Department to work in an effective and efficient manner.

In order to ensure that these processes are operating effectively and efficiently, it is necessary to measure their performance, assess whether they are adequately resourced, and improve them as required. To implement these requirements, process owners have been identified for each process. The process owner is a member of the Department's management team who has responsibility for the process and its results. Process owners draw on the expertise of the Departmental quality management team to support them in discharging their responsibilities.

Departmental oversight of the QMS is performed by a Departmental sub-committee chaired by a senior manager that meets regularly. The positions of the Departmental quality manager and divisional quality managers were created to support implementation of the QMS in a structured manner.

## 2. QMS tools

Within the scope of the QMS, new tools were developed and established to support the performance of Departmental processes so that Safeguards products achieve the intended purpose. These tools include continual process improvement, an internal quality audit programme, condition reports, knowledge management and a product-based cost model. The QMS tools are described in more detail below.

### 2.1 Continual Process Improvement

Continual Process Improvement (CPI) is a methodology and tool for identifying and quantifying process and administrative waste. The process change resulting from a CPI exercise is supported by a formalized set of documents so that the implemented improvement is sustained and is not a one-time fix.

The CPI concept was introduced in the Department at the beginning of 2006 by the Section for Process Design (former Section for Standardization) in the Division of Concepts and Planning. Process improvements are initiated by staff members involved in a process with agreement from the process owner.

Initial and advanced CPI training courses have been developed and presented to staff since 2006. The increasing attendance of these courses demonstrates the high interest from Safeguards staff on the opportunities provided by the CPI process.

The CPI system has contributed to the improvement of Departmental activities over the last years. Some of its achievements are:

- The development of a new shipping process from/to States to eliminate delays related to the receipt of equipment and samples;
- Implementation of new procedures for the transportation of recording media to the Safeguards Equipment Receiving Area and for the processing of information from recording media, shortening the cycle for returning recording devices for field use; and
- The frequency for backing up surveillance information on removable recording media (RRM) to the mainframe computer was changed from monthly to every two weeks. Thus the cycle time for returning RRM to the inventory was shortened. This change was part of a process improvement to resolve a removable recording media inventory shortage.

### 2.2 Internal Quality Audit programme

The Internal Quality Audit (IQA) programme is used to determine:

- The strengths and weaknesses of Departmental processes/activities;
- Whether the planned arrangements as described by Departmental procedures conform to internal and legal requirements, and, where applicable, to the requirements of ISO 9001;
- Whether actual practices conform to the planned arrangements; and
- Whether the QMS is effectively and efficiently implemented and maintained and that the Department's processes fulfill the requirements of the QMS.

The Department has invested in the IQA programme and has a pool of over 20 trained auditors available. Internal quality audits started to be conducted in 2006. Since 2008 the IQA programme has been supported by external subject matter experts invited by the Department to join the audit team where deemed necessary.

Thirty-one audits were performed between 2006 and 2012. Audit outputs are reviewed regularly and following the last round of audits some changes to the programme are gradually being introduced. The

main findings of the review have resulted in changes to the planning of audits – the scope of the audits is being scrutinized more closely to ensure it is more focused on the needs of the target area (combination of multiple layered versus broad-based, as appropriate). In addition, the use of external experts is being encouraged to bring a fresh perspective and an independent outlook to audit findings.

The IQA programme has a rolling two year cycle, with audit topics selected from a combination of requests (e.g. pre-audit in preparation for ISO 17025 accreditation of analytical techniques at the IAEA's Safeguards Analytical Laboratories) and as a result of senior management review. The IQA programme is flexible and can be adapted to incorporate new requests as a result of issues that arise during the two-year cycle.

Audit non-conformances are managed by the condition report system. Improvement opportunities identified in reports often lead to CPI activities.

### 2.3 Condition Reports

The Condition Report (CR) system is fundamental to the QMS. A CR is initiated by any staff that identifies a nonconforming or a potential nonconforming situation. A systematic approach is used to determine the root cause of the condition. Actions are then implemented to eliminate the cause and thus the occurrence of future nonconformities. The whole process is formalized and documented.

Originally called a Corrective Action Report (CAR), the system was implemented on a trial basis in 2006 with six CARs being initiated to investigate identified nonconformities. The CAR was renamed to CR in the second half of 2012 due to changes in the system. Severity levels, used for CARs/CRs classification, were redefined. Also, the classification 'minor' was implemented for events determined to be of low significance. A 'minor' CR is now processed through the CR system without a root cause determination. Nevertheless, a minor CR is tracked and evaluated for trends that may be an indication of a future risk.

Successes with the CR system are supported by increased use of CR procedures. During the last years it was observed that attendance at specific QMS training courses, in particular the CR sessions, increased. The progress made by the Department within the investigation and determination/ elimination of causes of radiological incidents and the support given by the CR system to the correction of weaknesses related to the shipping of radiological samples from field activities to Headquarters are concrete examples of achievements.

As with any other QMS tool, the CR system is still evolving. Within the set of improvements to be implemented in the near future is a new tracking system which will provide users with a friendlier interface to a database on reported incidents and their respective solutions.

### 2.4 Performance Indicators

Performance Indicators (PIs) are an important tool to enable the Department to determine whether it is delivering services in an effective and efficient manner in accordance with approved plans and to gain insights from the information to deliver better performance in the future.

A pilot study is being undertaken to introduce process performance indicators to the Department. This is another tool to support process owners and management in monitoring process performance. The concept is to start with a small number of PIs relating to key processes and to gradually extend and increase their number as experience is gained with its use. An internal workshop is being organized in 2013 to propose a methodology for identifying PIs and linking them to the appropriate Departmental strategies that they support.

### 2.5 Document Control

Control of documents and records is an important element of the QMS. Prior to the introduction of the QMS the Department had a limited document control system; it was therefore a natural progression to introduce QMS principles and expand the system to ensure that staff had access to the latest versions of documents and that these documents were approved for use by the Department.

A specific role - document coordinator - was introduced in every organizational unit of the Department to administer QMS controlled documents and records. The Departmental document coordinator is responsible for all activities related to the maintenance of a reliable set of approved documents for the Department. A procedure describes the Departmental QMS principles of document control:

administration of documents and records, roles and responsibilities, quality control checks and mandatory metadata. Metadata is an essential characteristic of a QMS controlled document. In addition to unique identification number of the document, the following metadata elements are mandatory: title of the document, type of the document (policy, terms of reference, procedure, guide, form, report, record, external document, etc.), document owner, document purpose, scope, security classification, version date and next review due date.

The Departmental quality management sub-committee regularly discusses documentation control issues and proposes improvements. Several guides have been produced to support staff with the preparation of documentation.

### 2.6 Document Manager

A specific software application named Document Manager (DM) was developed and implemented in 2011 for managing the quality of safeguards documentation. The application provides the following functionalities:

- Overview of all QMS controlled documents through a master list with detailed information about status, validity, classification, etc.;
- Access to a list with the latest changes to the inventory of procedures, guidelines, policies, such as new documents, new versions of existing documents and withdrawn documents in a given period of time;
- Searching capabilities for active documents using keywords or specific filtering criteria;
- Browsing active documents by Safeguards processes or through custom classification;
- Notifications when changes are made to active documents; and
- Off-line application for travelling staff.

Although the full content of security classified documents are only available to staff with appropriate access in the DM, the metadata (owner, version reason, purpose and scope, staff involved) is available to all staff.

The efficient implementation of the new QMS documentation control system is assured through regular training on the DM tool and regular meetings organized with document coordinators.

### 2.7 Knowledge Management

Knowledge management (KM) has been implemented as an integral part of the QMS. In 2008, a small Departmental team with expertise in the areas of operations, information management, training, and quality management began implementing the approved KM activities with two main objectives: (i) to improve the management of the critical knowledge leaving the Department with retirement or separation of staff members; and (ii) to improve the management of knowledge in the safeguards processes on a day-to-day basis – i.e. process knowledge (e.g. how to conduct an inspection) and knowledge resulting from the process (e.g. an observation made at a nuclear facility during an inspection).

The KM tools designed and implemented for supporting the retirement programme focus on helping the supervisor of the departing staff member to identify what knowledge needs to be retained and how to retain that knowledge. Typical tools/techniques are used such as documentation, seminars, question and answer sessions, and storytelling. With regard to day-to-day KM, a pilot project was completed in 2011 that used a structured methodology to analyze a business process from a knowledge perspective. The objective of the pilot project was to identify knowledge gaps that hindered the effectiveness and efficiency of the process itself, and the sharing of knowledge gained within the process.

Experience shows that using a small team for the design, development and implementation of knowledge management tools brings success as new ideas and concepts are implemented quickly. Also, working directly with the targeted customer (e.g. the retiring staff member's supervisor) to implement new ideas and concepts is demonstrating good results. With support from small KM teams,

the current efforts related to facilitating the identification and capture of important knowledge leaving the Department with retiring staff are producing very good results.

Other lessons learned indicate that the implementation effort is often underestimated and that work needs to be focused on actual problems encountered by staff members, rather than what the team thinks are problems.

### 2.8 Cost Calculation model

Up until 2009 safeguards implementation costs could only be estimated for safeguards effort in the field (based on person days of inspection and calendar days in the field). Verification and evaluation effort at Headquarters was not included because there was no method available to adequately estimate costs of Headquarters activities. This deficiency was reported in several external audit reports.

The Department selected a product cost model as the basis for its cost calculation methodology. The products are the outputs of processes defined in the Department's process framework. The model estimates the resources required to implement the Department's core processes. By accumulating and allocating appropriate shared costs for each product or category of product, the model is able to calculate the true cost of a product or product category. Products range from 90(a) statement statements to States - reporting the results of inspections - to annual implementation plans for small States (category of product). This model has been validated using actual data reported since 2009 — both quantities and costs expended from the regular budget allocated to the Department.

For each of the products identified for inclusion in the cost calculation model, process maps were developed through an extensive collaborative effort in the Department involving more than 200 experienced staff members. The maps are cross-functional process maps which detail the steps in each process and time estimates for carrying out the steps. These maps form the basis of process descriptions which have been developed for key processes under the direction of process owners in the Department.

The model has several applications, such as providing understanding and awareness about the cost of safeguards products; providing a consistent and transparent basis for the estimation of safeguards implementation costs by State; determining the resource implications of process changes and quantifying proposed process improvements; and supporting decision making through cost/benefit analyses.

The model has been reviewed and validated by an external expert on several occasions, and will be further refined, updated and validated as data become available in future.

### 2.9 Training

Training in the QMS was originally developed and conducted by external experts. The Department has subsequently developed its own training capacity and all QMS training is conducted internally as part of the Department's Safeguards Training Programme. Completion of QMS training is mandatory for all staff and is included as modules in safeguards training for new staff, including inspectors. QMS training covers recording of nonconformities and corrective actions, basic CPI, document control, internal quality audits and advanced QMS training.

## 3. Conclusions

QMS implementation is ongoing and evolving in the Department of Safeguards. The QMS infrastructure and tools are in place: the Department has a quality policy endorsed by the DDG-SG; Departmental oversight is provided by a Departmental sub-committee; a Departmental quality manager oversees all aspects of the QMS and reports directly to the DDG-SG; quality managers have been assigned in all organizational units; and training on QMS principles and QMS tools, described in this paper, is regularly provided to staff in the Department.

Lessons learned from implementing the QMS include the need for full management engagement and commitment, finding the right balance between centralized control of the QMS and decentralized implementation of the system, and to better communicate the value of QMS and its problem-solving tools to all staff.

# Overview of Next Generation Safeguards Initiative UF6 Cylinder Monitoring Project

**Karyn Durbin, National Nuclear Security Administration**
**Michael Whitaker, Oak Ridge National Laboratory**
**Ed Wonder, QinetiQ-North America**

***Abstract:***

*Just one of the thousands of cylinders containing uranium hexafluoride (UF6) moving around the world could be very useful to a proliferant country. Each of these large cylinders can contain up to two significant quantities of 235U. The National Nuclear Security Administration's (NNSA) Next Generation Safeguards Initiative (NGSI) has launched a five-year program to investigate a unique identification system for UF6 cylinders and to develop a cylinder monitoring concept that could be used by facility operators and the IAEA. The goal is to design an integrated solution beneficial to both industry and inspectorates that would improve cylinder operations at the facilities and provide enhanced capabilities for safeguards inspectorates to deter and detect both diversion of low-enriched uranium and undeclared production. The five-year program consists of 6 separate incremental tasks: 1) define the problem and establish the requirements for a unique identification (UID) and Monitoring System; 2) develop a concept of operations for the identification and monitoring system; 3) determine cylinder monitoring devices and technology; 4) develop a database to support proof-of-concept demonstration; 5) integrate that system for the demonstration; and 6) demonstrate proof-of-concept. Throughout the NGSI program, extensive engagement with industry stakeholders, regulatory authorities and inspectorates is essential.*

**Keywords:** IAEA; safeguards; UF6; cylinders

## 1. Introduction

Thousands of cylinders containing uranium hexafluoride (UF6) are transported around the world annually as part of nuclear commerce among facilities at the front end of the nuclear fuel cycle. If a proliferant state obtained one of these cylinders—containing either natural uranium or low enriched uranium—it would have as much as ~50 kg 235U. This is enough material, after further processing, to make two significant quantities of highly enriched uranium (HEU). While the International Atomic Energy Agency (IAEA) currently verifies that the material in these cylinders remains in peaceful use within non-nuclear weapon states, opportunities exist to further strengthen the current safeguards approach and increase the effectiveness and efficiency of safeguards. The National Nuclear Security Administration's (NNSA) Next Generation Safeguards Initiative (NGSI) has launched a five-year program to investigate a unique identification system for these cylinders and to develop a global cylinder monitoring concept that could be used by facility operators, national safeguards authorities and the IAEA. The goal is to design an integrated solution beneficial to both industry and inspectorates that would improve cylinder operations at the facilities and provide enhanced capabilities for safeguards inspectorates to deter and detect both diversion of low-enriched uranium and undeclared production. We believe that cylinder identification and monitoring is an essential element of enhancing both the effectiveness and efficiency of IAEA safeguards at these front end facilities and also at the state level.

## 2. Background

Concerns about cylinder identification and verification have grown over the last decade as international nuclear commerce has increased and enrichment technology has spread. An idea for an international database for UF6 cylinders was introduced in 2008 in a paper that discussed the potential benefits of monitoring the location of cylinders through the creation of an international database.[1] The following year, a joint paper by DOE, the IAEA, and URENCO proposed a program to develop and implement a world-wide system for identification of UF6 cylinders.[2] The authors stated that an identification method that could be independently used by national authorities and international inspectorates could greatly increase confidence in matching reported shipments and receipts, quicker and more reliable verification of declared cylinder inventories, and an improved capability for safeguards inspectors to assure that no undeclared cylinders are present at a facility.

It is important to understand the significance of UF6 cylinders, current operational practices for identifying cylinders, as well as what key stakeholders (i.e., industry and IAEA) are saying about these current practices and how they might be improved.

### 2.1. Significance of UF6 cylinders

The nuclear industry uses large metal pressure vessels called cylinders to store and transport UF6. The 235U concentration of the uranium most typically transported in these cylinders is either natural or low-enriched (i.e., <5%). A 2009 study estimated that there are approximately 20,000 of these cylinders in global circulation on an annual basis moving among conversion, enrichment, and fuel fabrication facilities.[3] Over the past 10-15 years, revelations about nuclear smuggling networks and clandestine enrichment facilities have focused attention on IAEA verification on the front end of the fuel cycle and on these cylinders in particular.

### 2.2 Current Operational Practices for Identifying Cylinders

Operators currently use a variety of methods to identify cylinders. According to a 2011 paper published by URENCO, operators use one of the following numbers inscribed on the cylinder nameplate for cylinder identification: 1) the manufacturer's serial number; 2) the owner's serial number; and 3) the National board registration number.[4] Because the nameplates can be difficult to read, operators have been known to mark the cylinders with stickers, stencils, or grease pencils to help identify the cylinders. It is not uncommon to see a cylinder with multiple markings including additional nameplates, multiple adhesive labels, painted markings, and bar codes.[5] Multiple and often conflicting markings make cylinder identification for both operators and inspectorates more difficult and time-consuming. A limited field test at a U.S. fuel fabrication facility in 2009 demonstrated that a conceptual, passive UF6 cylinder tracking system showed "potential to effectively track UF6 cylinders in a facility reducing the amount of human interaction needed, subsequently, reducing human error, and providing extra security for the accountability of UF6 cylinder."[6] Tests with a handheld reader showed that the time to take an inventory could be greatly reduced compared to conventional methods.

### 2.3 Key Stakeholder Perspectives

[1] Benefits of an International Database for UF6 cylinders, Babcock et al, 2008 INMM annual meeting, Nashville, TN

[2] "A concept for a world-side system of identification of UF6 Cylinders," Friend, Lockwood, & Hurt, Proceedings from the Institute of Nuclear Materials Management 50th Annual Meeting, 2009

[3] "Monitoring Uranium Hexafluoride (UF6) Cylinders," Eccleston, G. et al, Published by Oak Ridge National Laboratory, ORNL/TM-2009-128, June 2009

[4] "URENCO's Position on Standardising the Identification of UF6 Cylinders," Friend, Johnson, Engbers, Proceedings from the Institute of Nuclear Materials Management 52nd Annual Meeting, 2011

[5] Ibid

[6] Rose Martyn, An Operator Perspective from a facility evaluation of an RFID-based UF6 Cylinder Accounting and tracking System, 2011 INMM annual meeting

Industry representatives have recommended that an identification tag should be: unique; authenticatible; sustainable; acceptable to all stakeholders; suitable for full range of environmental and operational conditions; tamper-indicating; and secure.[7] Participants at an international conference for operators, inspectorates, and governments in Chester, England in 2009 agreed that "there needs to be an industry standard cylinder identification (ID) system for UF6 cylinders. There should be a global standard for unique ID numbering, a systematic and permanent way of marking the ID on the cylinders, and automated methods of reading cylinder ID's."[8]

One of the IAEA's key objectives in its Development and Implementation Support Programme for Nuclear Verification 2012-2013 is to "select and develop the techniques to uniquely identify the [UF6] cylinders supporting their tracking at facilities including those attached to the process."[9] The IAEA also notes that an international group of operators, regulators, and inspectorates agreed in 2009 that a global identification system should be undertaken by the community of operators (not the inspectorates) and therefore the "IAEA continues monitoring the maturation of identification/tracking technologies in support of possible safeguards use."[10] During the 2010 Safeguards Symposium, Agency representatives noted that "the large size, complexity and increasing automation within the process areas of enrichment plants has resulted in intensive efforts to identify measures to enhance safeguards approaches. The IAEA is studying alternatives to optimize its use of inspection resources while at the same time maintaining credible safeguards implementation."[11]

The potential benefits of a unique cylinder identification tag are summarized in Figure 1.

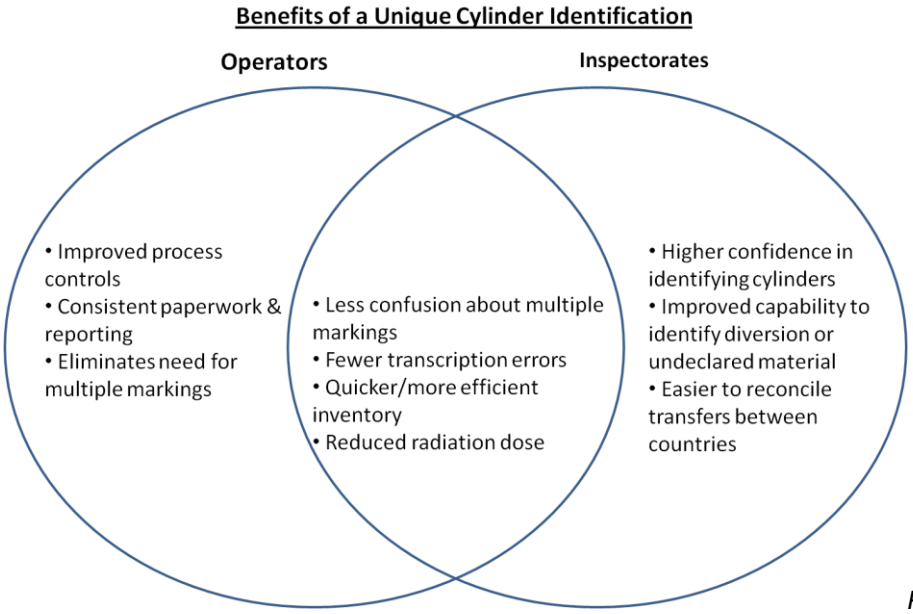


*Figure 1*

## 3. NGSI 5-year program

In April 2011, the NNSA NGSI launched a five-year program to investigate a unique identification system for UF6 cylinders and to develop a cylinder monitoring concept that

[7] "URENCO's Position on Standardising the Identification of UF6 Cylinders," Friend, Johnson, Engbers, Proceedings from the Institute of Nuclear Materials Management 52nd Annual Meeting, 2011
[8] "URENCO Conference on GCEP Safeguards, held in December 2009," Friend, Peter, Proceedings from the Institute of Nuclear Materials Management 51st Annual Meeting, 2010
[9] "Development and Implementation Support Programme for Nuclear Verification 2012-2013," International Atomic Energy Agency, STR-371
[10] Ibid
[11] "Laser-based Monitoring of UF6 cylinders," S. Poirier, M. Moeslinger, C. Liguori, D. Langlands, M. Burmester, International Atomic Energy Agency Safeguards Symposium 2010, IAEA-CN-184/213

could be used by facility operators and the IAEA.[12] The program was developed to be a systematic and structured approach to investigating solutions to cylinder identification challenges in ways that could benefit both the IAEA and operators. As such, the NGSI program was intentionally structured to ensure that the NGSI team methodically defined the challenges to be addressed before jumping to proposed solutions. The ultimate objective of the program is to demonstrate at a proof-of-concept level the principal elements of a global monitoring scheme that uniquely identifies UF6 cylinders throughout their life cycle. NGSI will engage the IAEA Safeguards Department as appropriate throughout.

The five-year program consists of six separate incremental tasks: 1) define the problem and establish the requirements for a unique identification (UID) and Monitoring System; 2) develop a concept of operations for the identification and monitoring system; 3) determine cylinder monitoring devices and technology; 4) develop a registry database to support proof of concept demonstration; 5) integrate that system for the demonstration; and 6) demonstrate proof-of-concept.

Task 1 was broken into several discrete sub-tasks. The first sub-task was to research and document the lifecycle of a UF6 cylinder. Building on the lifecycle analysis, the second sub-task analyzed probable diversion, undeclared production pathways, and related concealment strategies associated with UF6 cylinders at front end fuel cycle facilities. The third sub-task then analyzed how the IAEA would detect those diversion and undeclared production pathways based on current practices and how long such detection would likely take. In parallel with these sub-tasks, the team also engaged with key industry stakeholders including Westerman, USEC, ConverDyn, URENCO, Global Nuclear Fuels, Westinghouse, and AREVA. The final sub-task was to examine existing relevant national and international standards related to UF6 cylinders to understand what is currently required.

Task 2 also had several subtasks. The first sub-task was to develop and define the functional requirements for an identification and monitoring system based on the analysis that had been completed in Task 1. The other sub-tasks involved developing a concept of operations for an identification and monitoring system.

Task 3 will survey and evaluate technologies based on the functional requirements that were developed as part of Task 2. Following the technology survey, the team will down-select candidate technologies that are commercial-off-the-shelf or near-to-market technologies to be used as part of the proof of concept. The team will also complete a gap analysis based on the technology survey that can be used to inform future research and development decisions.

Task 4 will focus on the development of the information systems necessary to implement the identification and monitoring concept and more specifically to support the proof of concept.

Task 5 will integrate all the various pieces (e.g., technology, information systems, coordination with partners) to prepare for the proof of concept demonstration in Task 6.

The sixth and final task is the demonstration at a proof-of-concept level of an identification and monitoring system for UF6 cylinders.

## 4. Status

[12] "Next Generation Safeguards Initiative: Overview and Policy Context of UF6 Cylinder Tracking Program" Boyer, B. et. al, Proceedings from the Institute of Nuclear Materials Management 53rd Annual Meeting, 2012

As of April 2013, the NGSI team has completed Tasks 1 and 2.

### 4.1 Task 1: Defining the Problem

As part of Task 1, the team published a series of studies intended to help the team and the broader nuclear material management community understand the cylinder identification challenges and to lay the groundwork for future efforts. These studies were:

- "The Life Cycle of 30B and 48Y Cylinders" J. L. White-Horton, et al., ORNL/TM-2011/522, April 2011
- "Identifying UF6 Cylinder Diversion and Undeclared Production Pathways" S. Branney, et al., SRNL, January 2012 (not publicly available)
- "Methods of Detecting Diversion and Undeclared Use of UF6 Cylinders" M. Curtis, et al., PNNL-22017, November 2012 (not publicly available)
- "Developing an International Standard for Identifying Uranium Hexafluoride Cylinders," Carolynn Scherer and Brian Boyer, LA-UR-12-01286, LANL, February 2012

Once Task 1 was completed, the team produced a "Key Findings" report summarizing the major conclusions from the Task 1 efforts. These key findings included:

- National and international standards for cylinder fabrication have minimum requirements for marking and labeling cylinder; however, the standards do not currently specify a truly unique identification format that could be applied industry wide.

- The lifecycle-of-a-cylinder analysis provided important information about the environment within which a unique identifier and monitoring system would have to work, and also provided insights into how a unique identifier could be attached in a way that would survive a typical cylinder lifecycle.

- Given that there is no unique tamper-indicating cylinder identification tag, an inspector currently does not have a systematic way to detect certain concealment activities such as nameplate swapping.

### 4.2 Task 2: Identification & Monitoring Concept

Based on the lessons learned from Task 1, the team developed a concept for an identification and monitoring system based on three key elements: a unique identifier (UID) for cylinders; unattended UID reader systems; and a global registry of cylinders. The identification and monitoring system would focus initially on 30B and 48Y cylinders in active circulation. The application of a UID would be implemented over a 3-5 year timeframe as cylinders come due for recertification. Over the long-term, NGSI would work with the appropriate stakeholders on national and international standards to include the UID as part of cylinder fabrication requirements. More information on the NGSI concept for an identification and monitoring system can be found in an accompanying poster.[13]

### 4.3 Industry outreach

Once the framework for a conceptual monitoring system had been developed, the team held a series of meetings with industry stakeholders to gauge their reaction to the cylinder identification and monitoring concept. These meetings included representatives from USEC, URENCO, Cameco, Global Nuclear Fuels, Converdyn, and Westinghouse. The companies were generally receptive to the idea of standardizing a UID for all cylinders. However, there were the expected questions about cost burdens, application of the UID, and who is responsible for replacing the UID if it gets damaged. There were also some concerns about the "registry" (cylinder database) concept and the need to ensure that proprietary information be protected.

[13] "Preliminary NGSI Concept of Operations for a Cylinder Monitoring System," Whitaker, Michael, 2013 ESARDA conference

## 5.0 Path Forward

The NGSI program team is now moving into the third major task of the 5-year program which focuses on technology. Based on the feedback received from industry, the registry concept will be further analyzed to articulate the anticipated benefits of such a concept and address industry concerns. Stakeholder engagement will continue in the near-term including meetings with industry representatives, presentations at international conferences, and participation in other specialized meetings (e.g., U.S. Nuclear Materials Management and Safeguards System Users meeting). NGSI will be looking for tangible opportunities to work with industry leaders on cylinder identification approaches.

## 6.0 Conclusion

The NGSI mission is to develop the policies, concepts, technologies, expertise, and infrastructure necessary to sustain the international safeguards system as its mission evolves to meet new challenges. Accordingly, the NGSI cylinder monitoring program is focused on solutions that will help to strengthen the international safeguards system and make it more effective and efficient. Implementing systemic changes, such as the conceptual UF6 cylinder identification and monitoring system at nuclear facilities globally will be a long-term and multi-faceted undertaking. NGSI looks forward to working closely with all relevant stakeholders to continue to strengthen the safeguards system and ensure that it becomes more efficient without compromising its effectiveness.

# UK National Occupational Standards and Training Standards for Nuclear Material Accountancy and Safeguards

**Brian Burrows**

BNC Limited, United Kingdom
E-mail: Burrowsnc.ltd@virginmedia.com

***Abstract:***

*The UK is embarking on new nuclear build and with it a major emphasis on nuclear skills. The sector skills council for nuclear (Cogent), together with UK nuclear industry representatives have recently produced a set of national occupational standards and subsequently a set of industry training standards for those involved with Nuclear Material Accountancy, Control and Safeguards (NMAS). The standards have been developed to also be applicable to UK defence sites and to consider the nuclear security summit's challenge to improved interactions between nuclear security and nuclear material accountancy (NMA) in order to better detect unauthorised activities.*

*These standards apply to*

- *a wide range of personnel, both in the industry, contracted to the industry and those with oversight and governance of nuclear materials*
- *to the full nuclear fuel cycle; small and large nuclear sites, bulk and item processes, static and dynamic operations*
- *to all the stages of plant operation; design, commissioning, operation, and decommissioning*

*This paper describes the standards development, the challenges faced and how these documents fit in the national approach to skills. In addition these standards can be used directly by sites to benchmark their existing training materials and in a structured learning system to develop suitably qualified and experienced personnel. Lastly the document will consider qualification and recognition of the body of people who make up the NMAS profession.*

*The style of the document will be narrative form and take the reader through a personnel view of these developments. The author therefore does not claim to represent industry or national positions.*

**Keywords:** standards; training; NMAS; skills

## 1. Introduction

7th April 1975 was my first day at British Nuclear Fuels and working in a team to develop a new computerised nuclear material accountancy system. Now, at the end of my career, I am reflective about my lifetime in NMAS, a time of continuous learning with many exciting and challenging experiences. Safeguards, and its bedrock of NMA, are for many a technical discipline but for me the NMAS profession is much more. The people dimension is hugely important but in my experience it has not received the attention or the recognition it is due.

Those in the NMA community are seen as number crunchers and have little to define them as professional and capable. Compare this with financial accountants who are accredited by professional accountancy bodies against industry standards and ultimately reach chartered status – a sign that an individual has attained internationally recognised skills and competences.

Nuclear Material Control (NMC) also has no UK qualification and is often confused with nuclear security and nuclear safety. Nuclear material custodians tend to focus learning on gaining suitable experience within their immediate area of responsibility.

Are the skills required for NMAS simply based on a narrow set of needs, only relevant to the local context and only its particular part of the fuel cycle?

My experience tells me that there is a need for a much broader compass, a more holistic view of materials management and a wide understanding of the non-proliferation context and the needs for independent nuclear material verification.

In the safety world, the "people issue" requires constant emphasis on safety culture. Similarly in the NMAS world there must also be real attention paid to promoting a safeguards culture based on a sound and objective system of NMA, NMC and on human resources which are suitably qualified, experienced and reliable. These human resources often become the experts required to support international safeguards and non-proliferation initiatives and a resource on which the verification inspectorates can draw[1].

Figure 1 The Safeguards culture lexicon

At this point I want to dispel any impression that the UK has inadequate systems for training and competency development for its NMAS resources. On the contrary, each nuclear site tends to have its own training materials, highly tailored to the local context and often produced and delivered by NMAS experts. Such training is also accompanied by a company-wide human resource competency framework to assess and develop behavioural and technical competencies[2], and in a process context, is part of a system for ensuring provision of Suitably Qualified and Experienced Personnel (SQEP)[3]. However, a lack of commonality, the absence of national or international standards and the widespread differences in organisational structures and job content make it difficult to uniformly define NMAS skills. These factors together with the relatively small numbers of people involved mean that UK external training providers, exam/qualification boards and universities have not traditionally addressed NMAS.

## 2. The UK's nuclear portfolio

The UK nuclear legacy reaches back to the 1940's, with many of the nuclear licensed sites historically supporting both civil and UK defence requirements and with a number of nuclear sites dedicated solely to meeting defence requirements. The nuclear portfolio therefore covers the whole of the fuel cycle and its associated plant, equipment, systems, nuclear material assets and nuclear material liabilities (decommissioning and waste disposal).

There are over 30 nuclear licensed sites[4], covering power and research reactors; uranium conversion; fuel fabrication; uranium enrichment, spent fuel storage; reprocessing; plutonium recycle; waste storage and treatment plants; and a future which seeks new reactor build and provision of a geological disposal facility capable of taking spent fuel and nuclear wastes.

The UK also holds a considerable portfolio of civil fissile nuclear materials[5] on its nuclear sites including over a 100 tonnes of separate plutonium, over 1 tonne of highly enriched uranium and over 100,000 tonnes of lower grade uranium.

This wide range of facility types; company profiles; plant lifecycle points; material types and proliferation sensitivity show the challenging variability in NMAS job contexts in the UK.

## 3. A Community of Practice

The Nuclear Decommissioning Authority (NDA) was created in 2004 to manage the government owned assets of the UK civil nuclear industry. The NDA's remit includes requirements for knowledge management, skills retention and good practice dissemination and these are reflected in contracting arrangements for the management of its nuclear licensed sites. On 28th April 2008 I started work in the NDA as the Safeguards and NMAC Manager and worked in a team to develop better practices in Safety, Security and Safeguards[6]. This began with funding and organising industry workshops and conferences; by encouraging expert exchange; and by identification and promulgation of good practice.

One common technique for sharing is the formation of Communities of Practice (CoPs). Such communities help develop and advance thinking and in so doing nurture, elicit and disseminate knowledge. The NDA organised routine meetings of its site's NMAS personnel and invited attendance from others in the UK nuclear industry, including State authorities. In essence this was an NMAS CoP.

This wish to encourage interchange between individual practitioners also led to the formation of the INMM UK Chapter. That CoP would allow sharing with those involved with NMA and NMC in UK defence work, in UK private nuclear sector organisations and in UK academia. The NDA also promoted industry best practice sharing with WINS (the World Institute for Nuclear Security).

Whilst the NDA could resource it's CoPs, it could not mandate CoPs; the motivation had to come from the NMAS community and serve the real work of the sites. In the context of this paper, an important result of these CoPs was to complete an NMAS foundation guideline for the UK.

## 4. Guidance on International Safeguards and Nuclear Material Accountancy at Nuclear sites in the UK.

In 2006 I chaired a working group of the European Safeguards Research and Development Association (ESARDA) and in May 2007 I presented[7] the final reports of the Nuclear Material Accountancy and Control Audit Focus Group (NMACAF). This was at a time when the European Commission was refining it thinking on the use of audit by its Euratom safeguards inspectors.

**Figure 2 – ESARDA's NMACF Ad hoc focus group – Brussels, March 2007**

Sensible application of NMAC audit requires an appropriate reference to audit against. No ISO standard is completely congruent with the requirements of the Euratom Reporting Regulation 302/2005. The NMACF group consequently developed a reference guideline for good practice NMAC.

Quality Management measures the pulse of a business and considers many people factors such as leadership, communication, customer satisfaction, performance measurement, learning, continuous improvement, people satisfaction, commitment to a common sense of purpose, integrity and resilience.

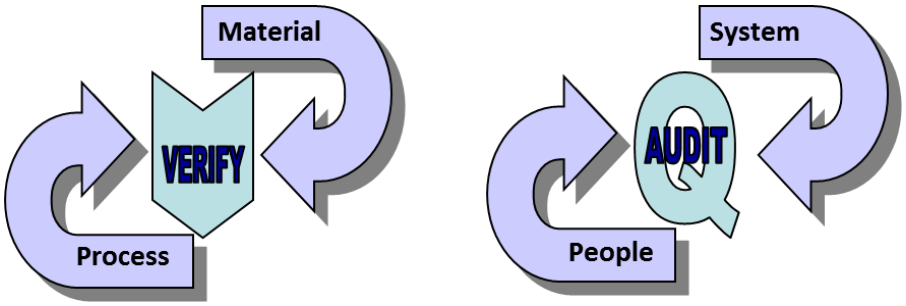


**Figure 3 NMAS- the people and quality management system dimension**

These potential developments in Euratom Safeguards' methods led the UK national safeguards authorities to initiate work to produce a standard or guide for all UK organisations holding nuclear material subject to safeguards, such that it could be integrated into the operator's quality management system. It became clear that such guidance would **not** be a formal UK standard (prescriptive, mandatory, and universally applicable for all ranges of users).

In 2009, NDA facilitated a complete redraft of the UK guide into a good practice style and to identify generic capabilities without mapping responsibilities to specific job roles. The guide was also enhanced to include more on capabilities for design for NMAS in new plants; measurement, sampling and analysis; NMAS during commissioning & decommissioning; and NMAS applicable to conditioned and unconditioned waste.

The NDA CoP played a key role in the development of the guide and interacted with the INMM UK chapter, British Energy and Urenco. The final version was sanity checked by seeking comments from Euratom and the IAEA safeguards inspectorates. The guide was benchmarked against a number of external guides, including the

- ESARDA NMACF guide
- European Commission guide (EC NMA recommendation c(2009)785 11/02/09)
- IAEA nuclear materials accounting handbook
- IAEA guide STR360 on designing for safeguards.

An agreed version[8] was finally issued in 2010 under the auspices of the UK Safeguards Office on the Health and Safety Executive (HSE) website.

## 5. The UK nuclear skills framework and the skills academy

The work on the NMAS guide came at the start of a nuclear renaissance in the UK against a wider backdrop of new reactor build driving a number of initiatives for better regulation, skills provision and training.

A National Skills Academy Nuclear (the Academy) was launched in 2008 to create, develop and promote world-class skills and career pathways to support a sustainable future for the UK nuclear industry. The Academy is one of the UK's family of Skills Academies and is also a subsidiary of Cogent, the Sector Skills Council for science based industries.

The Academy and Cogent established a database, called the Nuclear Industry Training Framework (NITF) to recognise the training and qualifications of relevance to the nuclear industry and industry

agreed role profiles called job contexts. The NITF provides the background information to the Academy's nuclear skills passport which accepts both formal and informal learning, meaning that it has the flexibility to recognise the training provided by individual employers, measured against national training standards. The nuclear skills passport recognises individual's achievements and their progress towards a competence profile based around job contexts.

The UK national skills framework includes National Occupational Standards (NOS)[9]. These have been established as a benchmark of competence required in the sector and can be used to develop training standards or vocational qualifications. NOS are produced as a suite of units for an occupational area and consist of a detailed breakdown of the tasks, knowledge and skills needed for effective performance and offer a framework for addressing good practice. NOS look at what needs to be achieved and are 'outcome' based.

National Occupational Standards provide the building blocks for nationally recognised qualifications and are also used by a variety of training providers to identify gaps in training provision. They are particularly useful for individuals to undertake a self-assessment of their competences against their own or other jobs. This can be helpful for someone considering a career move or simply for their own professional development. Employers can use NOS in a variety of ways such as:

- *Selection and Recruitment* - NOS can be used to form the basis of job descriptions and assist with staff retention
- *Staff Development* - NOS contain descriptions of good practice and can be used as the basis for setting objectives in performance and appraisal.
- *Developing and Evaluating Training* - Training plans and training courses can be developed to meet both organisational and individual learning needs. NOS can be used to inform the content of training as they specify what constitutes good practice. They can also be used to evaluate training by defining the expected outcomes.
- *Benchmarking* - The NOS provide an excellent basis from which to develop bench marking exercises either internally or to compare with other similar organisations.

NOS are intended to be UK-wide, and must represent an industry wide consensus on what it means to be competent in a given function. In order to achieve this, Cogent works with nuclear employers to develop standards that are fit for purpose and which meet the national criteria. Cogent is the custodian for its sectors' NOS and holds a prospectus list[10] of all the standards and qualifications underpinned by the NOS it manages.

## 6. Skills passport scheme

The Academy has established a network of providers through which these standards and qualifications are delivered. The Academy is also the lead organisation on the Skills Passport Scheme[11], used to manage skill requirements in a nuclear future where there is forecast to be an increased demand for skilled workers.

The passport will record and recognise individuals: achievements, training and skills development to industry agreed national standards and provide evidence of an individual's competency and skills. It will also record details of any site specific training taken at each nuclear licensed site. This will allow greater mobility of a skilled workforce throughout the industry and its supply chain.

The passport system will not replace a site's human resource system for competency assessing individuals nor will it replace the system for determining suitably qualified and experienced personnel (SQEP). However, it will provide a physical system for proof-of-skills, in particular for the contractor workforce.

## 7. Development of the NMAS NOS suite.

In 2010, I had the great fortune to meet Clive Smith, the Nuclear Director at Cogent[12]. During my time at NDA I attended some of our nuclear security CoP meetings. I met Clive at a CoP meeting on

Information Technology Security Standards and Competences where presented the subject of NOS to the meeting.

Given the publication of the NMAS guideline, it struck me that this was perfect timing for the formation of National Occupational Standards for NMAS.
Clive was very supportive, and soon Cogent was hosting an industry meeting to develop an NMAS NOS suite. The NMAS CoP which had produced the guideline was still in place and formed the core of the NOS working group. Great progress was made by combining the Cogent functional mapping process with a translation of the NMAS guideline into a set of NOS activities and outcomes.

Functional mapping is a three stage functional analysis: establish the key purpose; identify the main functions to achieve that key purpose; and identify what needs to be done to achieve each main function. Analysis must cover the whole of the NMAS occupational area and must align with the UK NMAS guideline. This gave rise to fourteen individual NOS units:-

1. Configure and manage the NMAS system
2. Define and deploy approved nuclear material measurement capability
3. Maintain and review nuclear material measurement quality control
4. Carry out nuclear material measurement system analysis
5. Identify and incorporate NMAS requirements in plant design
6. Confirm plant commissioning process achieves NMAS requirements
7. Identify and incorporate NMAS requirements in decommissioning plans
8. Control internal nuclear material movements on site
9. Control nuclear material movements onto/off site
10. Perform stocktaking and material verification
11. Resolve and investigate NMAS anomalies and discrepancies
12. Collect, collate, check and authenticate NMAS data
13. Compilation of accounts and NMAS reporting
14. Liaison with Stakeholders

Each unit has a title, followed by four sections: overview; performance criteria; knowledge and understanding; and additional information section. It must be easy for someone from the industry to see from the title what the NOS is about.

The performance criteria is a list of the key outcomes that whoever carries out the function must be able to deliver. Each performance criteria must begin with an action verb, and must make clear the objective/outcome and the context or condition. The process utilised lists of appropriate action verbs. The list below gives some action verbs but is by no means exhaustive.

| *Knowledge retention* | *Comprehension* | *Application* |
|---|---|---|
| List | Summarize | Solve |
| Name | Explain | Illustrate |
| Identify | Put into your own words | Calculate |
| Show | Interpret | Interpret |
| Define | Describe | Relate |
| Recognise | Compare | Apply |
| Recall | Demonstrate | Classify |
| State | Differentiate | Put into practice |

| *Analyse* | *Synthesis* | *Evaluation* |
|---|---|---|
| Analyse | Design | Evaluate |
| Organise | Hypothesise | Choose |
| Choose | Support | Estimate |

| | | |
|---|---|---|
| Contrast | Compare | Judge |
| Compare | Devise | Defend |
| Distinguish | Create | Criticise |
| | Construct | Justify |
| | Report | |

It is important to keep each NOS unit to a reasonable size to avoid a document that is quite large and probably would not be well received by the employers and staff members who will be using them. The majority of the NMAS NOS units consist of only one page of performance criteria and one page of knowledge and understanding. This is achieved by condensing blocks of descriptive and informative text into shorthand phrases which are then expanded in a glossary included in the additional information section. These shorthand phrase are bolded (where first used) to indicate that a full definition/description can be found in the glossary. Each NOS unit is standalone and self-contained.

The NOS units are deliberately worded so as to be suitable for use by civil and defence site personnel. UK safeguards arrangements do not cover nuclear sites used solely for defence purposes. On these sites, the UK Ministry of Defence (MOD) conducts material audits and independent verification and scrutiny of material accounts. The NOS units achieve this inclusivity by using the term “stakeholders” to encompass all independent verification requirements and by limiting the unnecessary use of Safeguards jargon.

The NOS units are also deliberately worded so as not to be job title specific. In small or static organisations an individual often carries out multiple NMAS functions whilst in larger organisations there is probably a greater division of labour and more specialisation. Each NOS qualifies the target audience in its overview and makes clear who the unit would primarily be for and who it may also be applicable to.

The NMAS NOS units were checked by the Cogent Education & Qualifications team and then forwarded to the UK Commission for Employment and Skills for lodging on the National NOS Directory. Cogent had to provide clear evidence of consultation (for them to be accepted as National Standards) and that all relevant parts of industry were consulted.

The consultation process included functional and training staff both in the civil and defence arenas. The wider safeguards and non-proliferation community were approached and comments received included feedback from the US Department of Energy; from ESARDA members; and from Euratom and IAEA safeguards inspectorates.

## 8. An example NMAS NOS unit

The finished NMAS NOS units were published[13] on the UK NOS standards database in January 2012 and are publically available (in pdf form) via the website of the UK Commission for Employment and Skills (UKCES). The best way to understand the content of a NOS is to view a real NMAS NOS unit.

### NMAS NOS unit - Perform stocktaking and material verification

#### Overview

This NOS forms part of a suite of standards which cover the activities carried out by individuals working within and on behalf of nuclear site licensed companies to meet nuclear material accountancy, control and safeguard (**NMAS**) requirements.

What is the NOS about?

A nuclear licensed site must ensure that nuclear materials are accounted for, controlled and safeguarded in order to demonstrate; good governance arrangements; meeting international safeguards commitments; and compliance with legal requirements and any voluntary undertakings. This NOS describes the standard expected of individuals who are responsible for physical inventory taking and verification as part of the NMAS system.

Who is the NOS for?

This NOS is primarily for Nuclear Material Custodians and NMAS Managers within nuclear site license companies who are responsible for compliance with NMAS requirements for stocktaking and material verification at a plant or site level.

The main outcome of this activity is the production of a verified inventory list of all nuclear material present in the inventoried area at a specified point in time.

Where text is highlighted in bold, it is more fully defined in the Glossary section of this NOS.

## Performance Criteria

*You must be able to:-*

1. comply with the **NMAS requirements** for Physical Inventory Taking (PIT) and conduct a PIT at least annually
2. arrange PITs to optimise inventory accuracy to minimise measurement uncertainty and communicate/notify PIT frequency and timings in advance to **stakeholders as required**
3. review and apply **PIT procedures**, adjusting preparations in line with conditions expected at time of the PIT and arrangements and information requirements discussed with stakeholders
4. identify and document any areas where a PIT is not possible and establish suitable alternative arrangements
5. identify and obtain required manpower and assign and communicate all roles and responsibilities to those conducting the PIT and those resolving the results
6. obtain approval and remove access barriers to allow PIT or Physical Inventory Verification (PIV) to commence
7. identify and process any outstanding transaction data in order to obtain up-to-date lists of inventory items from the NMAS accountancy system
8. transfer materials to designated locations, ensure correctly labelled and identified and, if necessary, take measurements
9. establish and document inventory items present and resolve any **discrepancies**
10. take samples and readings from process equipment to produce a list of material in process sufficient to assign nuclear material mass
11. obtain relevant technical justifications to underpin estimated hold up values
12. follow security arrangements to prevent theft/diversion during PIT, and report suspected loss or falsification of data
13. authorise and issue the results of the PIT
14. support Physical Inventory Verification (PIV) by independent stakeholders
15. hold the PIT position as agreed with stakeholders until their PIV has taken place. Resolve differences between PIT & PIV
16. ensure data provision sufficient to allow calculation of overall material balance uncertainty and action levels.

## Knowledge and understanding

*You need to know and understand*

1. the NMAS requirements, and **Supplementary Safeguards arrangements**
2. adequate **PIT/PIV capabilities and resources** appropriate to the required inventory taking/verification
3. packaging and container information (including tare weights), labelling and

identification systems and physical location maps

4. **associated regulatory requirements** for the nuclear materials which have to be considered at PIT and for the staff conducting the PIT (e.g. personal protective equipment)
5. the PIT procedures, reconciliation methods, and audit arrangements
6. the NMAS **implementation framework** for the areas being inventoried and for problem resolution
7. **process context**
8. plant nuclear material holding locations and access arrangements
9. inventory assessment techniques for active/heterogeneous materials including destructive analysis sampling plans, sealing technical justifications and non-destructive analysis
10. the tools and techniques used by stakeholders for independent containment, surveillance and inventory verifications
11. the barriers to verification activities

### Additional Information

*Glossary*

**Associated regulatory requirements:** such as Safety, Security, Waste Management, Environmental Protection, Transport and Import/Export Controls.

**Discrepancies:** include:

1. Differences between nuclear materials accounting information.
2. Differences in material balance.
3. Incorrect labelling of nuclear material packaging
4. Incorrect characterisation of nuclear materials
5. Nuclear material location errors

**Implementation Framework:** includes the NMAS physical and the managerial arrangements. It defines; the Material balance areas; transfer boundaries; key measurement points; NMAS capabilities, resources and infrastructure; control arrangements. It defines; organisational structures, responsibilities and accountabilities, separation of duties, those with direct custodial care of nuclear material and the competency framework

**Etc.**

## 9. Training standards for NMAS

Once the NMAS units were in place the next step was to put them to good use. The NMAS NOS working group had so far taken “small steps at a time” and with the advice of Cogent, chose to work next on creating NMAS training standards. These are a basis for training providers to structure their training courses but are also very valuable to nuclear sites for accrediting their own site based training as meeting national standards and as an objective way to assess NMAS competence. They also allow recognition of the training undertaken via inclusion on the national skills passport

Again the NMAS CoP took up the task and given the availability of the NOS units, then the training standards were produced quickly over a few months. Cogent again led the process, and was instrumental in guiding the group to a quick and successful outcome in a short number of meetings.

To ensure that training standards covered all NOS requirements, a mapping exercise was conducted relating each NOS performance criteria to its use in a training standard and each training standard assessment criteria to its NOS performance criteria.

The mapping exercise showed that some of the NOS concerning measurement were not fully covered by the NMAS Training Standards. The missing skill elements were more appropriate to the functional area of Metrology and Instrumentation and hence are deliberately not included in NMAS training. The mapping however flags to sites the need to ensure separate training provision in this area. Training in measurement systems is much more likely to be covered by external courses.

The training standards are aimed at two levels of training; an awareness level and a detailed practitioner's delivery level.

In the case of awareness standards, the action verbs are less forceful and address a broader audience for which a fairly short period of training is appropriate. The training delivery time for awareness standards is set as a minimum of half a day and the associated assessment methodology is largely to check understanding and the required learning outcomes.

Detailed delivery standards provide an in-depth knowledge and are expected to be deployed both in terms of formal learning and in terms of guided learning hours; again a minimum training time is suggested as one week. Each detailed training standard lists the NMAS NOS units to which it complies.

The NMAS training standards were finalised by the CoP in January 2013 and include:-

1. Awareness of Nuclear Material Accountancy and Control (NMAC)
2. Awareness of International Safeguards
3. Nuclear Material Accountancy (NMA)
4. Nuclear Material Custodianship (NMC)
5. Awareness of designing for NMAS
6. NMAS for transfers of Nuclear Material
7. Awareness of NMAS for decommissioning

Appendix 1: Link to National Occupation Standards

Appendix 2: Glossary of Terms

Appendix 3: Person Specification

Appendix 4: Recommended Reading

At this stage I think it is worth reiterating the purpose of the exercise to form NMAS training standards. The existence of industry training standards enables nuclear sites to map their site based training to recognised standards and, once accredited by the Academy, enables individuals to gain accreditation of their training on the Nuclear Skills Passport. It also enables sites to map their training to industry good practice, raising the bar on skills where necessary and provide regulatory confidence.

National Training Standards must go through a process of endorsement and approved. The UK nuclear industry has set up a skills advisory group (Nuclear Industry Standards Advisory Group) which reviews and endorses training standards. The Standards Advisory Group approves the standards once endorsed by industry. The training standards have been through the process of approval and will be made available to the UK nuclear industry and to approved training developers.

## 10. An example NMAS training standard –

### Nuclear Material Custodianship

### Learning outcomes:-

1. Identify the objectives of material control and custodianship within the context of your NMAS implementation framework

2. Illustrate how movements are controlled
3. Illustrate how Physical Inventory Taking (PIT) is conducted and controlled
4. Illustrate how anomalies and discrepancies are investigated and resolved
5. Explain how metrology systems used for nuclear material measurement are controlled
6. Describe the organisation and governance structures that exist for the management of NMAC (limit of authority, line of command)
7. Illustrate your PIT review and learning from experience processes

*Each of the above learning outcomes is given assessment criteria to fulfil. Taking learning outcome number 3 from the above the criteria are as follows:-*

### Assessment Criteria, the learner can:-

1. Describe the plant state and conditions required to optimise your inventory record accuracy, to minimise measurement uncertainty and to meet any specific requirements of external stakeholders

2. Explain how NMAS requirements for PIT are implemented in your quality management system

3. Explain how PIT resource logistics are managed and how roles and responsibilities are assigned and communicated. Explain how you prevent unauthorised activities (movements, substitutions, removal of material from batches/items, falsification of data) affecting PIT validity

4. Describe how you determine the amount of any material entrapped or held up in process

5. Describe how you determine the amount of nuclear material in your residues or in poorly characterised or heterogeneous materials such as wastes containing accountable nuclear materials

6. Explain how PIT findings are evaluated and reconciled with the accountancy books

7. Explain when a PIT is considered complete and operations can restart

8. Illustrate your PIT review and learning from experience processes

## 11. Job contexts

At large nuclear sites, there are a range of people who have an impact on the operation of the NMAS functional processes. These include operators; material custodians; project managers; designers; movement personnel; measurement personnel; IT specialists; commercial functions; and material accountants. At higher levels of management there are those responsible for issue such as governance, performance, culture, quality etc. Each of these populations has differing requirements for levels of training. However, two key groups of people within this population are material custodians and material accountants:-

- Materials custodians have direct control over material within their own plant area. Material custodians and deputy material custodians often form the largest part of the site's NMAS

population and are more likely to be mobile both on site and in transferring to other sites and nuclear companies.

- Material accountants provide the accountancy and regulatory reporting service for the custodians. Both are fundamental components of a site's nuclear material control and accountancy regime and interface with the independent verification stakeholders. Material accountants tend to be a static population and most likely to face demographic issues of retirement.

Cogent and the CoP have produced job context statements for these key NMAS roles. The nuclear passport is not however restricted to recording training solely against job context but will record any individual's training including NMAS awareness training.

## 12. Qualifications

In addition to developing training, NOS units can be used to develop qualifications. The process steps to achieve this are as follows:-

1) Decide what sort of qualification will be developed. This could be competency based and or knowledge based. Competence based is where the assessment would be based on observation of the activities in a real life working environment, along with questioning to establish the underpinning knowledge. Knowledge based is where delivery of the qualification would involve more of a formal training element, and assessment would be around learner's ability to apply knowledge to an assignment or via exam etc.
2) Engage with one or more awarding bodies to encourage them to offer the eventual qualification. Cogent has a route to do this through its awarding body forum, which meets quarterly. The forum would require data on likely take-up and market and would only support the qualification if student numbers are seen as viable for the effort.
3) Translate and agree the content of the NOS into qualification units based (as with training standards) on learning outcomes - what the learner can do as an outcome of the learning; and on assessment criteria - how to judge whether the learner has met the learning outcome.
4) Assign each unit credits and level values. A Credit - is a measure of the volume of learning which has been achieved, and the level is a measure of the complexity of the learning. Exercises would normally be used involving industry/assessment experts.
5) Develop a qualification structure, or 'Rule of Combination'. This sets out which units must be achieved in order to award the qualification. The rule can be configured in many different ways, traditionally this is based on mandatory units which all learners must achieve and optional units were learners can select from a wider choice.

The difficulty with a specialism such as NMAS is the limited number of people involved. Even in the UK, the numbers of those with clear responsibilities for delivering NMAS is relatively small. At this point, Cogent sought industry feedback on those numbers. Whilst many people have a need for good NMAS awareness, the numbers of NMAS practitioners would be unlikely to be seen as viable by awarding bodies for qualification development or by external training providers. Of course a qualification would have provided the NMAS discipline with a more professional feel, a better level of recognition by the nuclear industry, and a better reward for those who seek personal development. However, the production of the NOS, the training standards, and the job context make future formation of qualification units (in 2 above) a much simpler task.

Many years ago, during my time as the Head of NMAS in British Nuclear Fuels, my boss at that time, Dr Roger Howsley, the Director of Security, Safeguards and International Affairs asked me to consider how we could improve the professionalism of NMAS staff and how could they reach a level of competence at which they would be good "Ambassadors" for the Company; able to speak authoritatively in many forums.

That question came at a time when I was the Chairman of ESARDA and was partly addressed by encouraging NMAS staff to affiliate with such bodies and by my collaborating with the ESARDA Knowledge Management and Training working group on this issue.

That ESARDA working group led to the development of the ESARDA safeguards course, delivered each year in Italy, both to participants from the nuclear industry but also to students on nuclear degree courses. For those on degree courses, the safeguards' training is considered as a module attracting credits under the European Nuclear Education Network. Perhaps the issue of qualification will be met via universities, especially when new nuclear build significantly increases the pull for new resources.

At this point I should note that the Institute of Nuclear Materials Management (INMM) has also considered the need for development of a Safeguards and Security Professional Certificate Program. The INMM concluded that it found no strong drivers in the stakeholder community, particularly regulators; that educational institutions were again providing courses and certificate programs related to INMM disciplines; and that the cost of development and maintenance of a certification program by INMM would be prohibitively expensive.

I also note that WINS has produced a best practice guide on "Developing Competency Frameworks for Managers with Nuclear Security Accountabilities". Dr Roger Howsley, Executive Director for WINS said,

> *"We simply take it for granted that the certification and licensing of professionals indicates they have acquired the knowledge, skills and experience necessary to perform their jobs at a high level of expertise. It's what generates confidence.*
>
> *Unfortunately, in the realm of nuclear security, no certification requirements exist. Nor are there any professional development programmes recognised industry-wide as meeting the unique needs of managers with nuclear security accountabilities. In fact, there isn't even a generally-recognised profession titled "Nuclear Security Professional."*
>
> *WINS is now working hard to change this. Our ultimate goal is to enhance nuclear security leadership and professional development worldwide. The first step in this process is to create a competency framework that elicits the exact knowledge, skills and personal attributes managers with nuclear security accountabilities need in order to perform their jobs successfully.*
>
> *Once these competences are clearly understood, WINS will create professional development materials and certification opportunities to meet these needs and make them available through the recently launched WINS Academy. We have been privileged to work with international security practitioners and the National Skills Academy Nuclear in the UK to develop the Guide and through our liaison with the IAEA hope to promote this new and exciting professional framework".*

From this I take a hopeful view that NMAS can also have a new and exciting professional framework but perhaps the issue of qualification can only be cost effectively met at the International level. In the safeguards world this could be the IAEA acting as the qualification body certifying training providers as done in the UK by the National Skills Academy Nuclear.

Perhaps the answer to improve professionalism and qualification requires the presence of strong drivers in the stakeholder community. Consider figure three above, the answer lies with stakeholders taking more interest in the quality issue of systems and people. The ESARDA audit group concluded that to deploy audit, Euratom did not need further regulation but did require some benchmark against which to audit. This led to a European Commission guideline which addressed a benchmark for NMA systems. If NMAS NOSs could become international and used by safeguards inspectorates and state authorities to gauge operator competence and maturity then we would see more motivation for qualification.

## 13. Summary

I hope you have enjoyed my personnel view of these UK developments. The path to the NMAS NOS units and to the NMAS training standards has been a credit to all those who participated in the NMAS CoP and to Cogent, but it is now up to industry to benchmark their existing training materials and to develop the training to show how learning outcomes can be achieved. I think the UK, has at this point, a unique framework for addressing NMAS skills beyond the restrictions of being only adequate for the job on hand.

As a self-proclaimed NMA nerd I have always wished to dispel the occupational stereotype of the material accountant as a meek and tidy soul with a passion for numbers but performing what is simply a recording task. Functional mapping has shown the bigger picture and the full potential that can be reached within a framework of real delivery of outcomes and objectives.

For me the issue of skill is a matter of persistent achievement. The process of acquiring more knowledge is not necessarily skill and can make individuals unrealistically confident. Proper nuclear material management requires that competencies are more than just adequate and are based on proper objective assessment of capability.

## 14. Acknowledgements

A key enabler for this work has been Stakeholder support, particularly in funding resources to carry out the work. In this respect, I would like to acknowledge and thank the following:-

- The UK safeguards authorities for funding specialist support to develop the NMAS guide and the National Occupational Standards

- To Cogent for hosting and directing the work, dealing with the UK skills authorities and for funding specialist support to develop the training standards

- To the Nuclear Decommissioning Authority for its support of the NMAS Community of Practice and its work on good practice sharing

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# Implementation of Euratom Safeguards: internal and external state of play

**Paul Meylemans, Oscar Alique, Wolfgang Kahnmeyer, Christos Koutsoyannopoulos, Piotr Szymanski**
European Commission (EC)

***Abstract:***

*International nuclear safeguards are constantly evolving, both at political and technological level, and the European Commission's safeguards services are adapting to the arising needs and challenges. Continuous effort is invested in the improvement of effectiveness and efficiency of safeguards in the European Union. It identifies, develops and implements all possible means and measures for improving the quality and reliability of Euratom's verification results and conclusions. Euratom implements this at both internal and external level.*

*On the internal side, the Nuclear Safeguards directorate has decided to implement an Integrated Management System (IMS) for its activities. The system integrates the requirements of the Internal Control standards of the European Commission, complemented with the requirements of an ISO 9001 based quality management system (QMS). The IMS will allow for the continual improvement of nuclear safeguards processes taking into account the input of all actors. Its QMS components will ensure transparency of all processes so that EU Member States and the IAEA can have an appropriate level of confidence in the quality of the directorate's safeguards results. Moreover, the IMS includes the implementation of the latest international standards for the conformity assessment of the Directorate, ISO 17020 for inspection activities, ISO 17025 for measurements, and ISO 17021 for NMAC audit activities.*

*On the external side, the EC commonly implements safeguards together with the IAEA based on three safeguards agreements. With the introduction of IAEA integrated safeguards completed at the beginning of 2010, a revised framework of partnership approaches for the main types of facilities, complemented by a set of technical support documents, was developed and implemented. In light of the Agency's move to fully implement the State-level concept, the various elements of the New Partnership Approach (NPA), which was agreed in 1992, are being scrutinized to maximize the effectiveness and efficiency of common safeguards. The EC considers that the IAEA State-level concept might profit from the development of a complementary regional level concept or specific regional level factors.*

*The Senior Management of the IAEA and the EC agreed in June 2011 to establish an "IAEA/EC Reflection Group". The Group was tasked with reviewing existing cooperation arrangements between the IAEA and the EC to identify further possibilities for, and the conditions under which, enhanced cooperation and additional arrangements could allow the IAEA to make better use of the Euratom safeguards system. The High Level Liaison Committee in October 2012 recommended to use the Reflection Group report as guidance for the preparation of concrete proposals and charged the LLLC to investigate if all the elements of the NPA were implemented to their full extent. The HLLC also requested the Reflection Group to continue its work and look further into the areas identified in their report that go beyond the current NPA arrangements and consider them carefully in close cooperation with the legal services of both inspectorates.*

**Keywords:** SSAC; RSAC; IAEA, EC, EURATOM, IAEA, QMS, Reflection Group; NPA

## 1. Introduction

In the current economical context, public services like the European Commission are requested to be more and more transparent, fully accountable for their operations and able to demonstrate that they are using financial and human resources in the most effective and efficient way. This is not different for the European Commission's safeguards services. The Nuclear Safeguards Directorate has always been operating according to well defined procedures and methods, but it was considered necessary to go further than that and to implement a quality management system, based on international standards to ensure the overall quality of the activities performed and the services provided to stakeholders. Moreover, the Nuclear Safeguards Directorate also wishes to demonstrate its conformity with specific standards for organisations performing inspections, for measurements and for audit activities.

Euratom and the IAEA have a long history of cooperation in the implementation of safeguards. However, during the last two decades international nuclear safeguards has been constantly evolving due to the introduction of the Additional Protocol, the drawing of broader conclusions about States and the resulting Integrated Safeguards approach. More recently the evolution of the IAEA's safeguards concept into the State-level concept has paved the way to a more flexible implementation of safeguards inspections with the aim to combine effectiveness of the approach with a maximum of efficiency. A more flexible implementation of safeguards in the European Union can only go hand in hand with a higher level of coordination between the Euratom and the IAEA.

Under the State-level concept the IAEA wishes, amongst other aspects, to make full use of the effectiveness of State and regional authorities responsible for safeguards and place greater reliance on their support to the IAEA in implementing safeguards, while maintaining the IAEA's ability to draw its independent safeguards conclusions. How that could be done in the European Union, was investigated by a group of senior officials of the IAEA and the EC, known as the Reflection Group. The Group was asked to review existing cooperation arrangements between the IAEA and the EC and to identify further possibilities for, and the conditions under which, enhanced cooperation and additional arrangements could allow the IAEA to make better use of the Euratom safeguards system.

## 2. Integrated Management System

In November 2011, the Directorate for Nuclear Safeguards of the European Commission decided to implement an Integrated Management System to its activities. The main objectives for the system can be summarized in:

- Ensure compliance and further development of the directorate with European Commission rules regarding internal control.
- Enhancement of the confidence of nuclear safeguards actors in Euratom nuclear safeguards conclusions
- Alignment of nuclear safeguards activities with the latest international standards for conformity assessment activities.
- Implementation of continuous improvement cycles to nuclear safeguards activities.

## 2.1. ICS + QMS = IMS

The main elements of the Integrated Management System are the 16 Internal Control Standards of the European Commission and the standard ISO 9001 *Quality Management Systems – Requirements*. Moreover, the system includes three models developed by ISO for organizations performing conformity assessment activities. Concretely, ISO 17020 for inspection activities, ISO 17025 for measurements, and ISO 17021 for audit activities.

The Internal Control Standards (ICS) of the European Commission are based on the internal control model of COSO (Committee of Sponsoring Organizations of the Treadway Commission). The purpose of the ICS is to give assurance to the management that the resources are used in the most effective, economic and efficient way to achieve the organization's objectives. The 16 Internal Control Standards can be grouped in 6 building blocks, namely, mission and values, human resources, planning and risk management processes, operations and control activities, information and financial reporting, and evaluation and audit. The ICS foresee the implementation of the Activity Based Management that consists on the set up of objectives by the Commission Services and the continuous monitoring of the degree of achievement of these objectives by means of performance indicators. A risk management approach in order to control the potential factors influencing the objectives set by the Commission services is also implemented. Evaluation of the main expenditure programs as well as of the main legislative initiatives is an essential part of the ICS. The ICS are very well adapted to policy making and expenditure based services, but not so much to implementing services.

The mission of the Nuclear Safeguards Directorate of the European Commission is the implementation of nuclear safeguards in the European Union in order to make sure that nuclear materials are not diverted from their declared use and to comply with the requirements of Agreements concluded with Third States and international organizations. Thus, the activities performed are not primarily dealing with policy making, neither with expenditure, but are mostly implementing activities. Therefore, the process approach to management is more suitable to the activities of the Directorate. The process approach as defined in ISO 9001 consists in the identification of the different processes that form the system of the organization, in order to manage them to produce the desired outcome. In this framework a process must be understood as a set of interrelated activities using resources and managed in order to enable the transformation of inputs into outputs.

In order to implement the different elements of the Integrated Management Systems a team of four persons has been set up. The quality infrastructure is formed by a quality committee where all the five units are represented, and five different task forces dedicated respectively to internal control, process approach, inspection, measurements and NMAC audits

## 2.2. Safeguards process model

The first step in any organization intending to implement a quality management system based on the process approach is the identification of the different processes together with their interactions. In order to do so, a high level process map has been developed for the Nuclear Safeguards Directorate containing three main types of processes:

- **Management processes**, being the processes allowing for the governance of the organization as risk management, internal audits or strategic management.

- **Core processes**, being the ones allowing the organization to fulfill its legal obligations. They are related to two main categories:

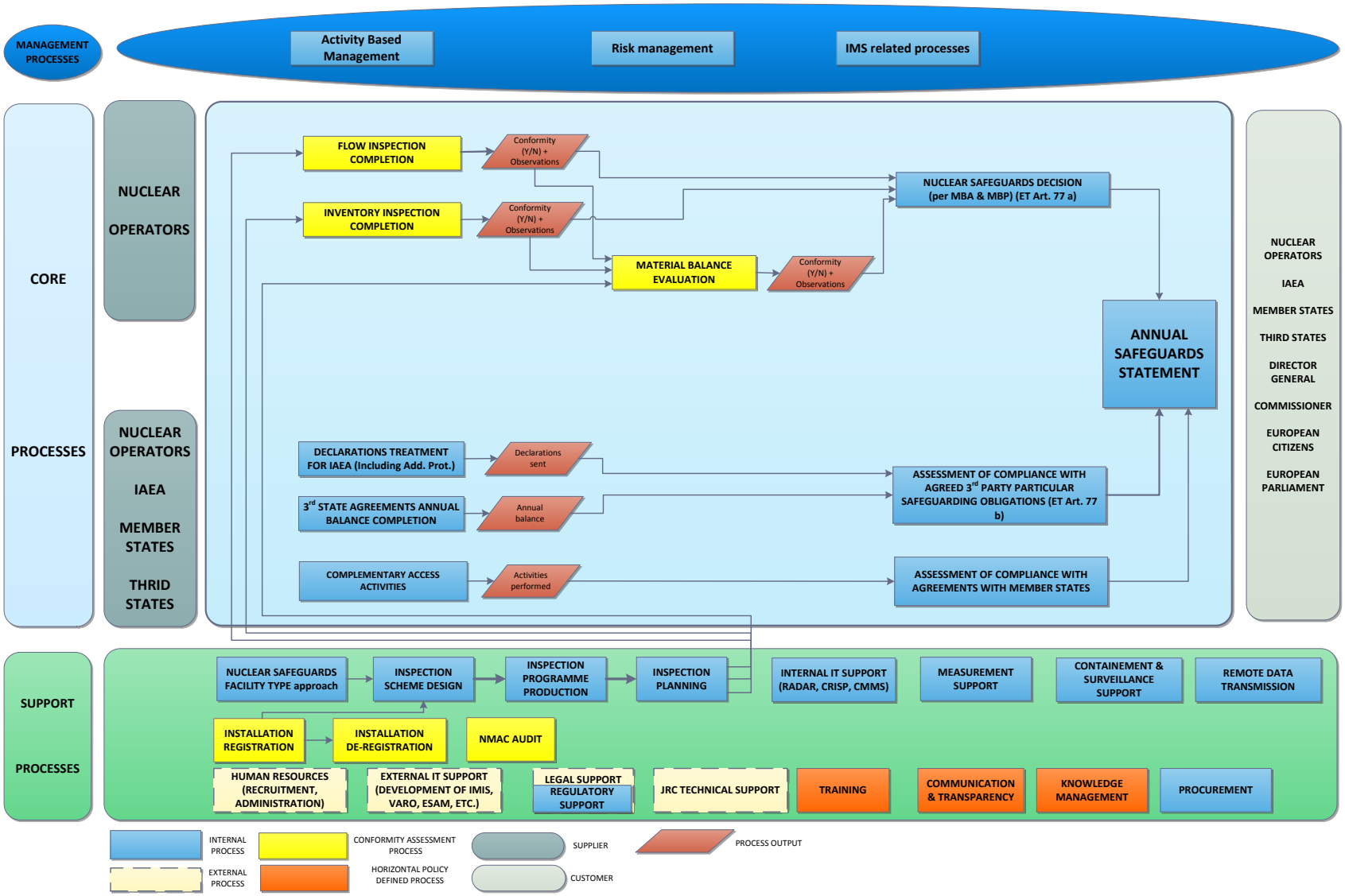
- The processes aimed at being satisfied of the non-diversion of nuclear materials (Art. 77a of Euratom Treaty) from declared uses. These are inspections of flows and stocks of nuclear materials, and material balance evaluations.

- The processes aimed at fulfilling the requirements of agreements with third states and international organizations (Art. 77b of Euratom Treaty) as the transmission of information to the IAEA in the context of the non-proliferation treaty, the other oblgations under the verification agreements with the IAEA including the processes related to the Additional Protocol, or the process to comply with agreements with suppliers of nuclear material (e.g. US, Australia, Canada etc.).

The products of these processes will allow the management to prepare annually a statement on the fulfillment of the legal obligations of the European Commission regarding nuclear safeguards.

- **Support processes**, being the processes that allow the organization to perform the core processes. Some examples are training, providing instrumentation and equipment, communication with stakeholders, or production of inspection schemes.

HIGH LEVEL PROCESS MAP. DG ENER E. NUCLEAR SAFEGUARDS



Once the main processes of the organization have been identified together with their interactions and sequences, they must be documented and mapped together with the subsequent sub-processes. At this point the risk management approach of the ICS and the set of objectives and key performance indicators will be integrated in the process approach. The management of the identified and documented processes includes monitoring the performance indicators and review of the objectives and the risk management of the process.

A documentation structure has been developed to allow the directorate to plan, perform and review the processes that form the management system. The documentation structure is divided in four categories with increasing level of detail. The documents are also divided into quality management documents and technical nuclear safeguards documents.

The technical document with highest level is called nuclear safeguards basic document and describes the legal framework of nuclear safeguards and how it is translated into operations. The second level is formed by the nuclear safeguards building blocks and the nuclear safeguards approaches. The nuclear safeguards building blocks describe the basic techniques used for nuclear safeguards. The nuclear safeguards approaches describe the activities to be performed by facility types using these building blocks. The third level reaches the operational category with the inspection scheme describing the activities to be performed in a material balance area during a material balance period.

### 2.3. Conformity assessment, measurement and audit

Conformity assessment is the demonstration that specified requirements relating to a product, process, system or body are fulfilled. Three important conformity assessment techniques are used in Euratom nuclear safeguards: inspection, testing (measurements), and audits of the nuclear operators' nuclear material accountancy and control (NMAC) systems.

The standard *ISO 17020 Conformity assessment – Requirement for the operation of various types of bodies performing inspections* has as objective the promotion of the confidence of the bodies that perform inspections. Inspections are the examinations of materials, products, installations, plants, processes, or work procedures to determine their conformity with stated requirements. According to this definition the main decisions made in safeguards belong to the framework of inspections. From the reception of the nuclear operators declarations in Luxembourg, through the control of supporting documentation, the review of measurement results and surveillance records, up to the verification of seals, most of the decisions made by nuclear safeguards inspectors belong to this category. Here, it needs to be kept in mind that the term inspection has a wider meaning in the context of IMS than normally used in safeguards where this term is generally understood as relating to verification activities performed on a nuclear installation site only.

The basic principle of ISO 17020 is putting together the right skills, the right resources and the right procedures in order to obtain the right decision about the conformity of an item inspection. The directorate documents its inspection procedures in a document called inspection scheme.

This document contains the activities performed in a material balance area in order to reach soundly based nuclear safeguards decisions at the end of a material balance period. The inspection scheme specifies the types and number of inspection activities to be performed, including the requirements of human resources and technical means. Moreover, the criteria against which the inspections are carried on are specified. Whenever necessary, working instructions specifying the way specific activities are to be performed will be drafted. The inspection schemes are subject to review and assessment regularly, and the inspection process is continuously monitored.

The inspection schemes also detail the measurements to be performed in nuclear installations. The standard ISO 17025 *General requirements for the competence of testing and calibration laboratories* specifies the requirements for bodies performing measurements and calibrations of measuring instruments. The inspection scheme will detail the conditions under which the measurements will have to be performed, the required level of accuracy, and the operational constraints to the measurement. A procedure has been put in place to determine what validated measurement method can be applied, and the measurement control measures to apply in order to ensure the metrological traceability and validity of the measurement results.

The NMAC systems of nuclear operators have an important role in the smooth running of nuclear safeguards processes. Audits of these systems are performed with the aim of improving nuclear safeguards efficiency. The standard ISO 17021 *Conformity assessment – Requirements for bodies performing audits and certification of management systems* specifies the way audits of management systems have to be carried out to find improvement opportunities. Audit questionnaires and procedures have been developed in line with this standard.

## 3. Cooperation with the IAEA in the implementation of safeguards

Based on the EURATOM Treaty, the European Commission (EC) commonly implements safeguards together with the IAEA in the EU based on three safeguards agreements [1].

The introduction of IAEA integrated safeguards (IS) for the non-nuclear weapon states (NNWS) of the EU was done as a step-wise process depending on the IAEA' capability to draw broader conclusions per country and to formulate their state specific approaches. This process was finalized at the end of 2009. To continue with a cooperative approach to the implementation of safeguards in the EU, previous inspection and technical support arrangements had to be adapted. As a result, a revised framework of partnership approaches for the main types of facilities together with a set of technical support documents was developed and implemented.

Some of these partnership approach and technical documents were updated during the last 2-3 years based on common implementation experience. This process will continue as needed.
In the light of the IAEA state level concept, the various elements of the New Partnership Approach (NPA), which was agreed in 1992, are also being scrutinized to maximize the effectiveness and efficiency of common safeguards implementation in the EU.

### 3.1. Coordination framework

Cooperation between the Euratom and IAEA inspectorates has been firmly built into the three safeguards agreements. They specify that safeguards activities under the agreements are carried out jointly by the Agency and Euratom and that the IAEA shall make full use of the Euratom system of safeguards. Different from the model agreement INFCIRC/153, INFCIRC/193 and the other two agreements therefore have a protocol attached which is an integral part of the agreement and amplifies certain provisions, especially on cooperation between Euratom and the IAEA.

The protocols provide that a High Level Liaison Committee between EC and IAEA meets once per year. It reviews the performance of the coordination arrangements and examines the development of methods and techniques. In addition there is a lower level liaison committee and working groups which meet more frequently. They consider, in detail, inspection procedures and their implementation, common inspection planning and technical support issues.

The existence of the committees and working groups provide a framework for continuous discussion and for the development and updating of procedures for the common implementation of safeguards in the EU.

### 3.2. Partnership Approach documents

Starting in 2008, in the process of introducing IAEA integrated safeguards in the NNWS of the EU, Partnership Approach (PA) documents for all major types of facilities were prepared:

- light water reactors,
- spent fuel storage facilities,
- research reactors and critical assemblies,
- gas centrifuge enrichment plants, and
- depleted, natural and low-enriched uranium conversion and fuel fabrication plants.

They describe approaches and inspection activities which allow both the IAEA and EURATOM to fulfil their respective obligations under INFCIRC/193 and INFCIRC/193/Add.8 and are adapted to reflect the IAEA's integrated safeguards and the European Commission's strategy for improved safeguards effectiveness.

In addition, facility specific partnership approaches were developed for the major bulk handling facilities (all enrichment plants and fuel fabrication plants) and for selected other facilities (on-load reactors). It is expected that the last few remaining documents of this kind will be finalized in mid-2013.

The above PA documents are fully replacing a large set of similar documents that were established in the mid-1990'ies as a result of implementing the "New Partnership Approach" (NPA) that had been agreed between the Director General of the IAEA Dr. Blix and Commissioner Cardoso e Cunha in 1992 [2].

The facility type and facility specific PA documents can be seen as a comprehensive set of documents for the common implementation of safeguards. They have partly been updated based on implementation experience gained in the last 2-3 years. Now, they are considered not to need any substantial changes in the foreseeable future.

On the technical support side, a number of new technical support documents were agreed which deal with technical aspects that became important as a result of recent technical developments or due to necessary changes with respect to previously established support processes.
New technical developments that needed to be dealt with are, for instance, the installation by the EC of surveillance equipment for common use, the arrangements for joint-use remote data transmission from nuclear facilities in the EU to both IAEA and Euratom headquarters, or the introduction of EOSS seals for common use. Another example of this kind are the arrangements for spent fuel cask verification and sealing in which the conditions for cask sealing by operators alone are an important aspect. Changes to previously established procedures relate, for instance, to the sealing arrangements for joint-use surveillance equipment or to surveillance review.

Currently, a "Joint Use Arrangement for the New Generation Surveillance System (NGSS)" is in preparation, where all relevant activities and practical arrangements from the camera keying, set-up, installation, transport and troubleshooting up to the review activities are described.

Related to support two more aspects are worth mentioning:

- Based on earlier NPA arrangements, revised cooperation arrangements for the training of IAEA and EC staff were established in 2008. Since then new types of common training activities like training in teamwork and communication and training on Partnership Approaches and the legal framework were introduced.
- Being aware of the responsibilities of both organizations for the safety of their staff during inspections on site, joint arrangements for radiation Protection, health and safety of inspectors were agreed.

A number of previously agreed arrangements covering areas like environmental sampling or joint analytical capabilities are currently being reviewed. It is expected that this process will be finalized towards the end of 2013.

### 3.3. Rolling Plan

The generic IAEA guidelines on IS include the use of randomly scheduled inspection as a standard element of the inspection scheme for many types of installations and give preference to the use of unannounced inspections. As long as such inspections are also unannounced to the EC they are to some extent contradicting to some basic principles of the verification agreement, especially the concept of wide cooperation and coordination of activities. It was therefore important for the EC to find ways to reduce the number of facilities where unannounced inspections are carried by the IAEA alone as much as possible. This was achieved through the use of surveillance or other technical means that allowed the introduction of random inspections with short notice for those facilities. This mode of implementing random inspections allows the EC to make the maximum use of random inspections for the attainment of Euratom inspection goals while also keeping the safeguards related burden on operators at a minimum.

From the beginning of 2010, a rolling scheme of inspections is used which provides for the regular transmission by the IAEA to the EC of advance information of random interim inspections with short notification. The early provision of this information gives the EC the opportunity to plan the resources required and to make arrangements for full participation. The information provided by the IAEA however does not identify the facility or State concerned, but merely gives the number of inspections planned in a particular time envelope in the future, thereby keeping randomness and unpredictability of IAEA inspections. Currently about 100 installations in 14 NNWS of the EU are covered by the rolling scheme. Annually, around 50 random inspections with short notice are carried out in these installations in accordance with the IAEA state specific requirements. Notification times are either 24 hr or 48 hrs (based on retention times at fuel fabrication facilities, or availability of surveillance with remote data transmission and other facility specific conditions at LWR, for example).

The rolling scheme can now be considered as a fully established and reliably working tool for the common implementation of random inspections in the NNWS of the EU.

### 3.4. Review of the NPA arrangements

The NPA of 1992 had developed further the general cooperation and coordination requirements which are part of INFCIRC/193, the safeguards agreement between all NNWS of the EU, Euratom and the IAEA.
Subject to the ability of both organizations to satisfy the requirements of their respective criteria and guidelines, the NPA requires the EC and the IAEA to implement a number of conceptual elements, especially:

- Replacement of the principle of 'observation' (as provided for in INFCIRC/193) by the one-job-one-person principle supplemented by quality control for all inspection activities while avoiding unnecessary duplication of effort
- Use of common safeguards approaches and inspection procedures
- Common inspection planning
- Commonly shared analysis capabilities for DA samples
- Use of common instruments, methods and techniques
- Cooperation in research and development
- Common training of inspectors
- Increased common use of technologies to replace physical presence of inspectors by appropriate equipment.

When IS started to be introduced in the EU about 5 years ago, the High Level Liaison Committee (HLLC) between EC and IAEA confirmed that the NPA and all its elements continue to be fully valid. There would however be the need to adapt the various detailed implementation arrangements that had earlier been agreed between both organizations. As mentioned above, this has been achieved in various areas through the elaboration of a number of partnership approach documents.

Keeping in mind that the NPA had its 20th anniversary last year and that, after a first review done in 1996, no further complete review of the NPA had been carried out thereafter, the HLLC in October 2012 decided to initiate a new review. This would aim at making detailed proposals for areas where and how the NPA can be more fully implemented. The review process has started and is closely connected to the work of the Reflection Group which is further detailed in another part of this paper.

### 3.5. Development of the IAEA state level concept

The IAEA has made various statements on the State-level concept (see for example [3]) mentioning, among others, the expanded use of State-specific factors and the importance of cooperation with regional and state systems of accountancy and control of nuclear material (R/SSAC).

The current status of evolution of the State-level concept appears however limited too strictly by an almost exclusive use of acquisition path analysis to establish State-specific technical objectives and their prioritization. This obviously results from the wide experience that the IAEA has gained mainly through the application of traditional safeguards measures but restricts the concept to purely technical features and parameters.

There are however other factors that may be used to further optimize the use of resources and thereby free some of the limited resources to be used in areas of highest safeguards concern. One such potential factor certainly is the proven record and technical capabilities of R/SSAC in the implementation of safeguards in a state or group of states. Another factor could the proven commitment and support given by a state or group of states to international non-proliferation.

It may however not be possible to properly consider the specific features of the regional Euratom system of safeguards within the current framework of the state-level concept and state evaluation process. An additional regional level concept could support and complement the state level concept of the IAEA. The application of a state level concept combined with a regional level concept could lead the IAEA to a regional level approach that makes better use of the Euratom safeguards system. [4]

## 4. Reflection Group

As a fundamental requirement of the safeguards agreement between the Community, the NNWS of the EU and the IAEA, the IAEA and the EC have always been looking at possibilities to improve the efficiency of safeguards activities in the EU, while keeping or enhancing their effectiveness. It is therefore not surprising that the senior management of the IAEA and the EC agreed in June 2011 to establish an IAEA/EC Reflection Group. The Group was tasked with reviewing existing cooperation arrangements between the IAEA and the EC and to identify further possibilities for, and the conditions under which, enhanced cooperation and additional arrangements could allow the IAEA to make better use of the Euratom safeguards system. The idea was in a first phase to investigate the strategic areas of enhanced cooperation between the IAEA and the EC and the underlying foundations that would allow such cooperation.

Four senior officials of both organisations were appointed as members of this Reflection Group and the group met on 6 occasions between July 2011 and March 2012, when they issued a report entitled "Proposal of the IAEA/EC Reflection Group".

The report identifies the underlying foundations considered to be necessary for enhanced collaboration between the EC and the IAEA, which would allow better use to be made of the Euratom safeguards system. These underlying foundations mainly relate to the competence of the Euratom safeguards system and its independence from nuclear operators or Member States. It is therefore considered that an organisation that implements safeguards should demonstrate its adherence to quality criteria by implementing a quality management system based on the general requirements that are described in ISO 9001, but also on the general criteria for the operation of various types of bodies performing inspection outlined in ISO 17020. The quality management system should allow other organisations that want to make use of service or of data, to perform activities to confirm that these are of the agreed level of quality. Such activities could be the inspection of procedures, data, records and reports and/or the performance of audits at headquarters or during on-site inspections.

In the report, the Reflection Group further identifies five strategic areas which are expected to have the most impact on collaboration between the two inspectorates and which offer the possibility to implement certain actions at an early stage. These areas are:

- An ISO based quality management system for Euratom safeguards, including establishing synergies with the IAEA Safeguards Department's QMS.
- Safeguards activities in the EU Nuclear Weapons States.
- The more efficient management and use of common instruments, processes and procedures.
- The broader application of the principle "one job, one person".
- The exchange and utilisation of safeguards information.

More detail on the outcome of the work of the Reflection Group during the first phase and which medium to long term cooperation goals are envisaged can be found in the paper on "Enhanced IAEA/EURATOM Cooperation and Reinvigoration of the NPA" presented in the 35th ESARDA Annual Meeting in Brugge.

The High Level Liaison Committee in October 2012 recommended to use the Reflection Group report as guidance for the preparation of concrete proposals and charged the LLLC to investigate if all the

elements of the NPA were implemented to their full extent. The HLLC also requested the Reflection Group to continue its work and look further into the areas identified in their report that go beyond the current NPA arrangements and consider them carefully in close cooperation with the legal services of both inspectorates.

Since then, the Reflection Group has been preparing Terms of Reference and a Guidance document on the application of the New Partnership Approach (NPA) between the IAEA and the Euratom Community for the application of safeguards pursuant to INFCIRC/193. The Terms of Reference define in more detail the issues that the Reflection Group will be dealing with and how it will coordinate its activities with the LLLC. The purpose of the Guidance document is to draw the attention of staff in the IAEA Division of Operations C and in the EC's Directorate for Nuclear Safeguards to the possibilities for enhanced cooperation introduced with and offered by the NPA.

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# Fault Tree and Event Tree assessment for security system

**Marko Čepin*, Dmitriy Demyanyuk**, Alexandra Sitdikova****

* Faculty of Electrical Engineering, University of Ljubljana
Trzaska 25, 1000 Ljubljana, Slovenia
**Institute of Physics and Technology,
National Research Tomsk Polytechnic University
Lenin avn.30, 634050 Tomsk, Russian Federation
E-mail: Marko.Cepin@fe.uni-lj.si, Demyanuk@tpu.ru,
Alexandra.Sitdikova@gmail.com

***Abstract:***

*Standard methods, presented by different researchers in the field of security systems assessment, are very complicated for implementing them nowadays. The purpose of this paper is to provide simple and applicable method for quantifying the efficiency of the system. We suggest implementing Fault Tree and Event Tree assessment, which is widely used way for evaluation of nuclear safety systems reliability, to design successful security system and to improve the existing one. The exploitation conditions plays a huge role in any system success, thus, we used special Common Cause Failure method to predict all possible failure events. Also the human error was taken into account to show how personnel actions could affect on failure/success of the system. In fact, proposed method could be applied for any kind of complex industrial system.*
*A standard security system was selected as a case study to present the advantages of proposed method. The results of this analysis could be used to check and improve weaknesses of the system, or to design the system with minimum probability of failure.*

**Keywords:** security system; fault tree; common cause failure

## 1. Introduction

Successful and fail-free interaction between elements, equipment and other technical means is the main issue in the design of effective physical protection system (PPS). The correct interconnected work mainly defines the success of the whole system's operation in general. PPS is required to make timely detection, to find an unauthorized activity for the effective adoption and implementation of the respond forces to neutralize the intruders. Elements of the PPS should have high rate of efficiency, reliability, solidity, etc. Nowadays the role of complex technical systems as a part of PPS is constantly rising.

A lot of attention is paid to the efficient construction and operation of the security systems for nuclear and non-nuclear facilities. To improve the reliability of the systems' components and equipment of the physical protection system the methods of probabilistic risk assessment can be successfully applied.

## 2. Probabilistic Risk Assessment

Probabilistic Risk Assessment is a systematic technique for investigating the transformation of an undesired initiating event into a set of possible outcomes and their consequences. For each initiating event, an event tree (ET) is constructing to follow a sequence of events through multiple stages of safety systems to be activated. For a safety system comprising multiple components, a fault tree (FT) is constructed to represent the combinations of component failures that could result in the system failure, treated as the top event of the tree. The top event probability of the FT is then obtained by combining, through Boolean algebra, failure probabilities of the components treated as basic events.

Because a probabilistic analysis is normally done for a highly reliable system, the probability of the events occurring usually can be determined with the rare-event approximation. Thus such a technique is best at identifying weaknesses or vulnerabilities in a system [1].

## 2.1. Common Cause Failures

PRA is a systematic process for examining how engineered systems, built and operated based on these requirements and practices, and human interactions with these systems work together to ensure plant safety. Typically all important facilities are design with redundant safety systems, redundant trains and redundant equipment for improved reliability and safety. However, very high reliability theoretically achievable through the use of redundancy is often compromised by single events that can individually render redundant components unavailable [2]. Such events are known under the term common cause failure (CCF) events or common cause failures.

CCF events have been recognized as the dominant contributors to the results of the system reliability analysis and PRA. They are defined as a subset of dependent failures in which two or more component fault states exist at the same time, or in short time interval, and thus they represent failures resulting from a shared cause [3]. For example, environmental CCFs include orbital debris strikes and exposure to excessively high humidity, temperature or vibration. Neglecting contribution of common caused failures can result a significant underestimation of risk. [4] CCFs are being acknowledged as one of the most challenging issues in the PRA, especially within PRA fault tree modeling of safety systems within nuclear power plants.

## 2.2. Human Errors

Just as there are elementary approaches for incorporating CCFs into a PRA, the Markov method can be used for treating human reliability in systems with only few components. Such an approach typically requires that a constant human error rate be assumed and hence does not adequately address the complexity or the operating conditions of a nuclear plant.
There are two types of human/nuclear plant interactions that need to be considered: those involving routine plant operation, testing, and maintenance and those involving plant safety issues. The two categories differ in the amount of stress an individual may be under. Because nuclear power safety systems are designed to operate automatically during the initial stages of an accident sequence, human intervention normally would not be required. In instances, however, where it would be required, the human response undoubtedly would be done under stress [5].
There are also two general categories of errors involved in human reliability analysis (HRA): errors of omission and errors of commission. Errors of omission involve actions in which operators take no action or where a set of actions taken leads to no significant difference between no action and the actions taken. Errors of commission involve actions taken that can significantly increase the severity of an undesired incident.
Three objectives of a human reliability analysis are:

- Identify sources of human error and human failure modes to be included as human failure events (HFEs) in a PRA framework or model.
- Develop models in the PRA representing the specific HFEs of interest.
- Quantify the human error probability (HEP) associated with each HFE, including understanding the factors that may most influence the HEP estimate.

## 3. Physical Protection System Design

Whether designing a new PPS or upgrades to an existing system, the designer must determine how best to combine such elements as fences, barriers, sensors, procedures, communication devices, and security personnel into a PPS that can achieve the protection objectives. A system may be defined as a collection of components or elements designed to achieve an objective according to a plan. The ultimate objective of a PPS is to prevent the accomplishment of overt or covert malevolent human actions. Typically objectives are to prevent sabotage of critical equipment, theft of assets or information from within the facility, and protection of people.
A PPS must accomplish its objectives by either deterrence or a combination of detection, delay, and response. These functions must be performed in this order and within a length of time that is less than the time required for the adversary to complete his task. A well-designed system provides protection-in-depth, minimizes the consequences of component failures, and exhibits balanced protection.

For a system to be effective, there must be notification of an attack (detection), and then adversary progress must be slowed (delay), which will allow the response force time to interrupt or stop the adversary (response).

Detection is the discovery of an adversary action. It includes sensing of covert or overt actions. To discover an adversary action, the following events need to occur:

1. A sensor reacts to a stimulus and initiates an alarm.
2. Information from the sensor and assessment subsystems is reported and displayed.
3. A person assesses information and judges the alarm to be valid or invalid. If assessed as a nuisance alarm, detection has not occurred. Therefore, detection without assessment is not considered detection. Assessment is the process of determining whether the source of the alarm is due to an attack or a nuisance alarm.

Detection functions in a PPS are shown in the Figure 1.

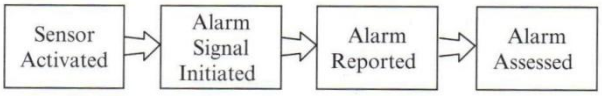


**Figure 1.** Detection Functions in a PPS

The measures of effectiveness for detection function are the probability of sensing adversary action, the time required for reporting and assessing the alarm, and nuisance rate. A sensor activates at time $T_0$, and then at a later time a person receives information from the sensor and assessment subsystems. If the time delay between when the sensor activates and when the alarm is assessed is short, the probability of detection, $P_D$, will be close to the probability that the sensor will sense the unauthorized action, $P_S$. The probability of detection decreases as the time before assessment increases. Figure 2 shows that a long time delay between detection and assessment lowers the probability of detection, because the more time required to make an accurate assessment, the less likely it will be that the cause of the alarm is still present. In addition, the delay between detection and assessment favours the adversary because of the further progression of the adversary toward the target before the response force has been notified of an attack.

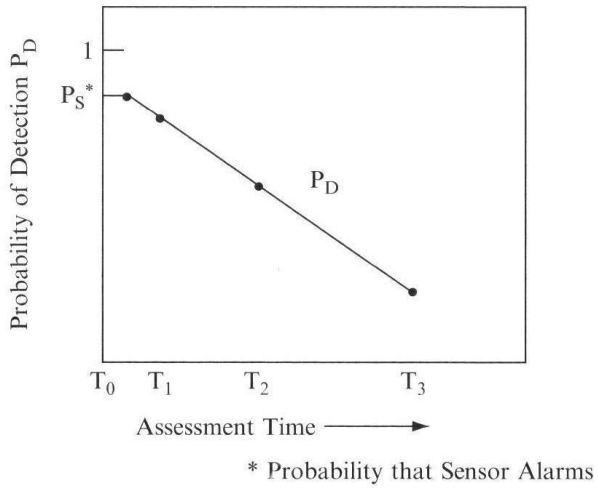


**Figure 2.** Ratio between Assessment Time and Probability of Detection.

## 4. Modified Alpha-Factor Method

Nowadays there are several analyses for estimating the vulnerability of the PPS. In this work the main goal was assess the reliability of the system using PRA technique with taking into account exploitation conditions, possibility of terrorist attack, operator errors and others. Different methods and techniques were studied, but there was no possible solution of simultaneous assessments for all-factors-including analysis. Thus, we modified standard CCF method for that purpose.

This paper presents the method based on Alpha Factor method, but applying for explicit modeling of single and multiple components failure events simultaneously within number of several different Common Cause Failure Groups (CCFGs) – sets. Each CCFG is defined on the basis of specific

coupling mechanism. All sets could be sorted by the group size – k (number of elements in it) and by the number of common elements in each CCFG, – x (if it is single than x = 1 or multiple x ≥ 1; but in any case x ≥ k). The presented method that accommodates components failure events to be simultaneously assigned to different CCFGs given different coupling mechanisms is based on a modification of the well-known Alpha Factor model. The motivation to develop this method was the incapability of one of the most widespread PSA software for FT and event tree ET modeling, [5], for simultaneous assignment of neither one single component failure event, nor multiple components failure event in more than one CCFG within the fault tree analysis technique. Namely, the software package provides with a CCF modeling feature based on manual assignment of arbitrary failure events, i.e. basic events (BEs), to specific CCFG upon selection of proper parametric CCF model. In the process of this assignment of BE to CCFG, the software does not accommodate the option for one to assign one BE to several different CCFGs, a scenario quite probable in practice since given component can experience failure due to different causes, which if seen as shared causes couple the specific component with other components in different CCFGs simultaneously.

The method was applied on a selected case study system. The application of the method enables improved PRA models. The improved models consequently implicate better results.

The method is based on traditional Alpha Factor Method with few modifications because of expansion for multiple CCFGs defined for a system. The extension goes by number of groups in every set and number of common elements. In parallel system of n trains, the function of one is enough for the system success, so all trains should fail for the system failure, i.e. the failure criteria is n out of n.

The probability of a set failure, concerning Boolean logic, could be calculated as a sum of all combinations of two, three, n element failures and independent failure of this element. Thus, the probability of failure for m-set Ps with single common component in it is calculated as follows:

$$P_s^k = P_{ind} + m\sum_{n=2}^{k} C_n^k P_{nCF} \quad (1)$$

where $C_n^k$- number of combinations n of k, $C_n^k = \frac{k!}{(k-n)!n!}$.

Ps – system failure probability;

Pind – independent failure probability of a component;

PnCCF - probability of n component failures due to common cause.

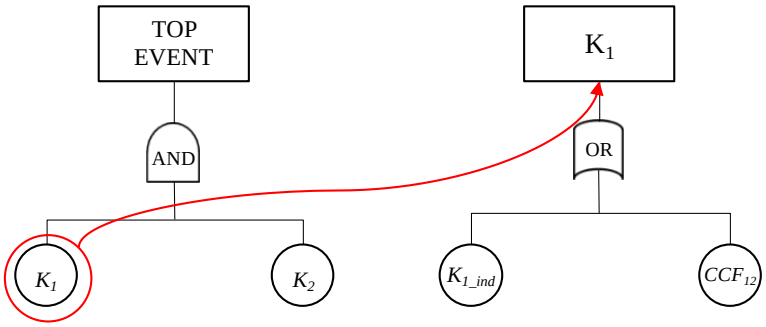


**Figure 3.** Component level (left) and sub-component level FT via explicit modelling (right)

Figure 3 shows the general idea of explicit modelling for CCF.

One more way of applying modified Alpha Factor method is to use the sets with more than one common element in CCFGs. The probability of multiple failure events in multiple common cause failure groups is calculated in this case. The probabilities of the basic events involving n specific components in a CCFG of size k (1 ≤ n ≤ k) can be defined as:

$$P_{nCF} = \frac{a_{n/k}}{a_t} P_k \quad (2)$$

This method is proposed for cases, where the set contains the groups with similar types of components in one train (i.e. diesel generators, check valves, motor pumps), where all component failure probabilities are equal. For component groups of different types of components, where several

failure probabilities are used, the mean value of $a_{n/k}$ normalizing factor $a_t$ and failure frequency of each component $P_k$ are calculated as follows:

$$a'_{1/k} = \sqrt[n]{\prod_n a_{1n/k}},\ a'_{2/k} = \sqrt[n]{\prod_n a_{2n/k}}, \quad (3)$$

$$a'_t = \sum_{n=1}^{k} n a'_{n/k}, \quad (4)$$

$$P_k = \sqrt[k]{\prod_{n=1}^{k} P_n} \quad (5)$$

So the probability of common cause failure involving n specific components in CCFG can now be written as:

$$P_{nCF} = \frac{a'_{n/k}}{a'_t} \sqrt[k]{\prod_{n=1}^{k} P_n} \quad (6)$$

Proposed method has several differences from standard Alpha Factor method. It can be seen as its upgrade. The main advantage of the discussed model is fact that implementation of such CCF approach could be especially useful for CCFG within several numbers of different types of components. Examples include number of system trains with components placed in one room or on the same floor, the number of similar components of the same producer with the same physical and technological characteristics.

## 5. Case study: the security alarm system

A model of Alarm System KODOS was chosen as a case study. The Alarm System is designed to detect attempts and/or the fact of unauthorized access; to inform the security personnel about these events, and automatically send the necessary commands to the actuators.
The case study consists of two parts: PRA analysis with calculating the general probability of system failure is the first part. The second part of this study presents CCF analysis with using proposed extended Alpha-Factor method, which taking into account such events as sabotage, extreme exploitation conditions, etc. and show its real impact in the system reliability. Thus, using proposed method we could estimate probability of system failure due to “outside” impacts, including intruder attack.
The first step was to analyse the principal scheme of Alarm system and to build the FT and ET for further assessments. Then, using Risk Spectrum PSA these trees were analysed, and the list of Top-10 events for system failure was obtained. Table 1 shows Top-10 events for PRA without CCF analysis. The probability of system failure amounted Q = 3,04E-04.

| No | Probability | % | Event | Top Event probability Q = 3,04E-04 | |
|---|---|---|---|---|---|
| 1 | 1,000E-04 | 32,79 | OS_FAIL | | |
| 2 | 1,000E-04 | 32,79 | COMMUN_FAIL | | |
| 3 | 1,000E-04 | 32,79 | SW_FAIL | | |
| 4 | 1,000E-06 | 0,33 | A-06/8 | | |
| 5 | 1,000E-06 | 0,33 | A-07/8 | | |
| 6 | 1,000E-06 | 0,33 | AKP | | |
| 7 | 1,000E-06 | 0,33 | P1_FAIL | | |
| 8 | 1,000E-06 | 0,33 | RCD_F | | |
| 9 | 1,000E-08 | 0,00 | P2_FAIL | P3_FAIL | |
| 10 | 1,000E-12 | 0,00 | L1 | L2 | L3 |

**Table 1:** Top-10 events for system failure without CCF analysis.

By the next step CCF analysis was applied. The FT and ET were modified due to Extended Alpha-Factor method. Each element were described with all possible failure events, such as lack of power,

serial fault / defect, failure to weather reasons, burglary / damage / sabotage, hacking through PC, installation error, etc. Then elements were grouped into several CCFGs. The failure due to weather reasons (humidity, too high/low temperatures, etc.) combined elements into CCFGs, and common event due to installation error and sabotage. Some elements have been presented in both groups, which helped to assess the contribution for each failure event.

The list of Top-16 events for system failure is presented at the Table 2. The probability of system failure in this case amounted Q = 7,21E-04.

| No | Probability | % | Event | Top Event probability Q = 7,21E-04 | |
|---|---|---|---|---|---|
| 1 | 1,00E-04 | 13,87 | OS_FAILURE | | |
| 2 | 1,00E-04 | 13,87 | SW_FAILURE | | |
| 3 | 8,50E-05 | 11,79 | AD-01_IND | | |
| 4 | 8,50E-05 | 11,79 | A-06/8 | | |
| 5 | 8,50E-05 | 11,79 | A-07/8 | | |
| 6 | 8,50E-05 | 11,79 | P1_IND | | |
| 7 | 8,50E-05 | 11,79 | AKP | | |
| 8 | 8,50E-05 | 11,79 | RCD_IND | | |
| 9 | 4,65E-06 | 0,64 | CCFG_II | | |
| 10 | 2,49E-06 | 0,35 | CCFG_III | | |
| 11 | 1,00E-06 | 0,14 | COM_LINE_FAIL | | |
| 12 | 1,00E-06 | 0,14 | COM_EV | | |
| 13 | 5,78E-07 | 0,08 | CCFG_V | | |
| 14 | 4,10E-07 | 0,06 | CCFG_IV | | |
| 15 | 1,00E-08 | 0 | P2_FAIL | P3_FAIL | |
| 16 | 1,00E-12 | 0 | L1 | L2 | L3 |

**Table 2:** Top-16 events for system failure with CCF analysis.

Marked events show the impact of CCF into the system reliability. CCFG_II has the highest number, the probability that two elements of the system fail due to wrong exploitation conditions is 0,64%. The probability of failure because of intruder attack is 0,14%. The probability of failure of four elements simultaneously has the lowest number, 0,06%.

Difference in numbers of top event probability (without CCF Q1 = 3,04E-04, with CCF Q2 = 7,21E-04), shows the impact of CCF into the system reliability, and underestimation of those factors could cause serious consequences.

## 6. Conclusion

Probabilistic Risk Assessment has a widespread approach for analysing nuclear safety systems. Although PRA can be applied to any engineered system. Typically the PPS is not analysed with PRA techniques, but implementing of proposed CCF method could significantly simplify the security systems vulnerability and reliability analyses. Using modified Alpha-factor method, almost all possible events could be taken into account for assessment: from extreme exploitation temperatures and high humidity, to earthquakes and human errors.

Another step in PRA for security systems could be implementing of Dynamic FT for vulnerability assessments. This would help to estimate correct actions of operator of alarm communication and display subsystem during the attack. Since only fast and correct actions let the PPS's function – Detection, to prevent the terrorist attack, human reliability analysis would get a great approach in assessments.

## 7. Acknowledgements

Prof. Dr. Marko Čepin supervised the most of these studies. He helped to develop the extended Alpha-factor method for CCF analysis. The Slovenian Research Agency supported this research (project J2-2182).

## 8. Legal matters

I agree that ESARDA may print my name/contact data/photograph/article in the ESARDA Bulletin/Symposium proceedings or any other ESARDA publications and when necessary for any other purposes connected with ESARDA activities.

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# Ontology-based semantic information technology for safeguards: Opportunities and challenges

**Michael McDaniel**

International Safeguards & Technical Systems
Sandia National Laboratories[1]
Albuquerque, NM, USA
mmcdani@sandia.gov

***Abstract***

The challenge of efficiently handling large volumes of heterogeneous information is a barrier to more effective safeguards implementation. With the emergence of new technologies for generating and collecting information this is an issue common to many industries and problem domains. Several diverse information-intensive fields are developing and adopting ontology-based semantic information technology solutions to address issues of information integration, federation and interoperability. Ontology, in this context, refers to the formal specification of the content, structure, and logic of knowledge within a domain of interest. Ontology-based semantic information technologies have the potential to impact nearly every level of safeguards implementation, from information collection and integration, to personnel training and knowledge retention, to planning and analysis. However, substantial challenges remain before the full benefits of semantic technology can be realized. Perhaps the most significant challenge is the development of a nuclear fuel cycle ontology. For safeguards, existing knowledge resources such as the IAEA's Physical Model and established upper level ontologies can be used as starting points for ontology development, but a concerted effort must be taken by the safeguards community for such an activity to be successful. This paper provides a brief background of ontologies and semantic information technology, demonstrates how these technologies are used in other areas, offers examples of how ontologies can be applied to safeguards, and discusses the challenges of developing and implementing this technology as well as a possible path forward.

Keywords: safeguards; information; ontology; semantic technology; knowledge management

## 1 Introduction

The effective application of International Safeguards requires that the IAEA assess large volumes of information to come to high-confidence conclusions regarding States' uses of nuclear materials and technologies. Compiled from state declarations, inspection activities, material accountancy, laboratory analyses, sensors, open sources, and more, this information is heterogeneous in format, diverse in content and distributed in space [1]. To compound the challenge of handling such information, the emergence of the State Level Concept requires that the information analysis cycle be continuous and integrated; conducted within the context of all available information including existing knowledge in order to develop a complete picture of each State's nuclear fuel cycle activities.

The information challenges faced by the IAEA in carrying out its safeguards mandate, however, are by no means unique. Indeed these are the same problems faced by many industries whose main asset is information. The IAEA and its supporting organizations can look towards these industries and disciplines for best practices and technologies to make the handling of information more efficient.

Semantic technologies are among the solutions being developed to handle heterogeneous information. These technologies provide a structured means of describing information so that the

[1] Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000. SAND Number 2013-3943 C

information to be understood and processed by computers in meaningful ways. Often these technologies rely on ontologies which represent explicit specifications of knowledge within a domain. Semantic information technology has great potential to increase the efficiency of safeguards by helping to integrate, organize, and analyze heterogeneous information, but significant challenges remain for these benefits to be realized. This paper describes ontology-based semantic information technologies and their potential role in addressing the challenges of information management for International Safeguards.

## 2 Sematic information technologies

Information resources are largely designed and formatted for presentation to humans rather than for automated processing and manipulation by computers. While there are examples to the contrary, such as well-designed relational databases, these are inflexible and based on specific, often narrowly defined, data models. Semantic information technology attempts to shift this paradigm by allowing information to be encoded in such a way that computers can interpret and processes the information, thus taking some burden off the human consumer of information.

The concept of semantic technology is not new. From the inception of the World Wide Web, Tim Berners-Lee envisioned a web of information that was meaningful to computers, rather than a web of documents meant primarily for human consumption [2]. However, by creating an information architecture that provides information that computers "understand" it is possible to shift much of the analytic workload from humans to computers.

While semantic information technology encompasses a diverse collection of technologies such as natural language processing, data mining and artificial intelligence, this paper focuses on the technologies and specifications that serve as the foundation of the Semantic Web. Note, however, that Sematic Web technology does not pertain only to document exchange over the Internet; these concepts are equally applicable to all information systems.

One of the fundamental components of semantic technology is the concept of ontologies. Ontology comes from the metaphysics branch of philosophy and concerns the study of the nature of reality and its contents. Computer scientists have borrowed this concept to address the challenges of sharing information between systems based on an agreed upon understanding of reality. In computer science an ontology is a formal specification of knowledge within a given domain [3]. Ontologies explicitly encode concepts, their properties, relationships among concepts, and the logic for how these relationships are defined. Ontologies are defined using formal language in order to allow knowledge to be reliably reused. In a similar manner to object-oriented programming, once a concept is defined within an ontology as a Class it can be used to create specific instances, or individuals, based on this Class. For example, the Class <Country> may be used to instantiate the individual <Belgium>. The instance <Belgium> will inherit all the properties of class <Country> (e.g. having a population, spatial extent, etc.). These individuals can also be encoded into the ontology.

The development of ontologies for a given domain has several benefits including creating a shared conceptualization of knowledge that facilitates the sharing of information between systems and people. A well-developed ontology can serve as the basis of many advanced information management capabilities. Below, the fundamental building blocks of semantic information technology are described.

Ontologies can be encoded in any number of formal languages. The *Web Ontology Language* (OWL) is a family of formal language specifications and the W3C recommended specification for authoring ontologies. OWL provides formal semantics and vocabularies for defining classes, their properties and their relationships, for defining individuals based on these classes and asserting their properties, and the logic necessary to reason about classes and individuals [4]. In this way OWL can be understood as a modeling language capable of creating very specific, descriptive and complex models of the knowledge within a domain. These models are specified using standards that make the information "understandable" to computers, allowing computers to manipulate the information described by the model in meaningful ways.

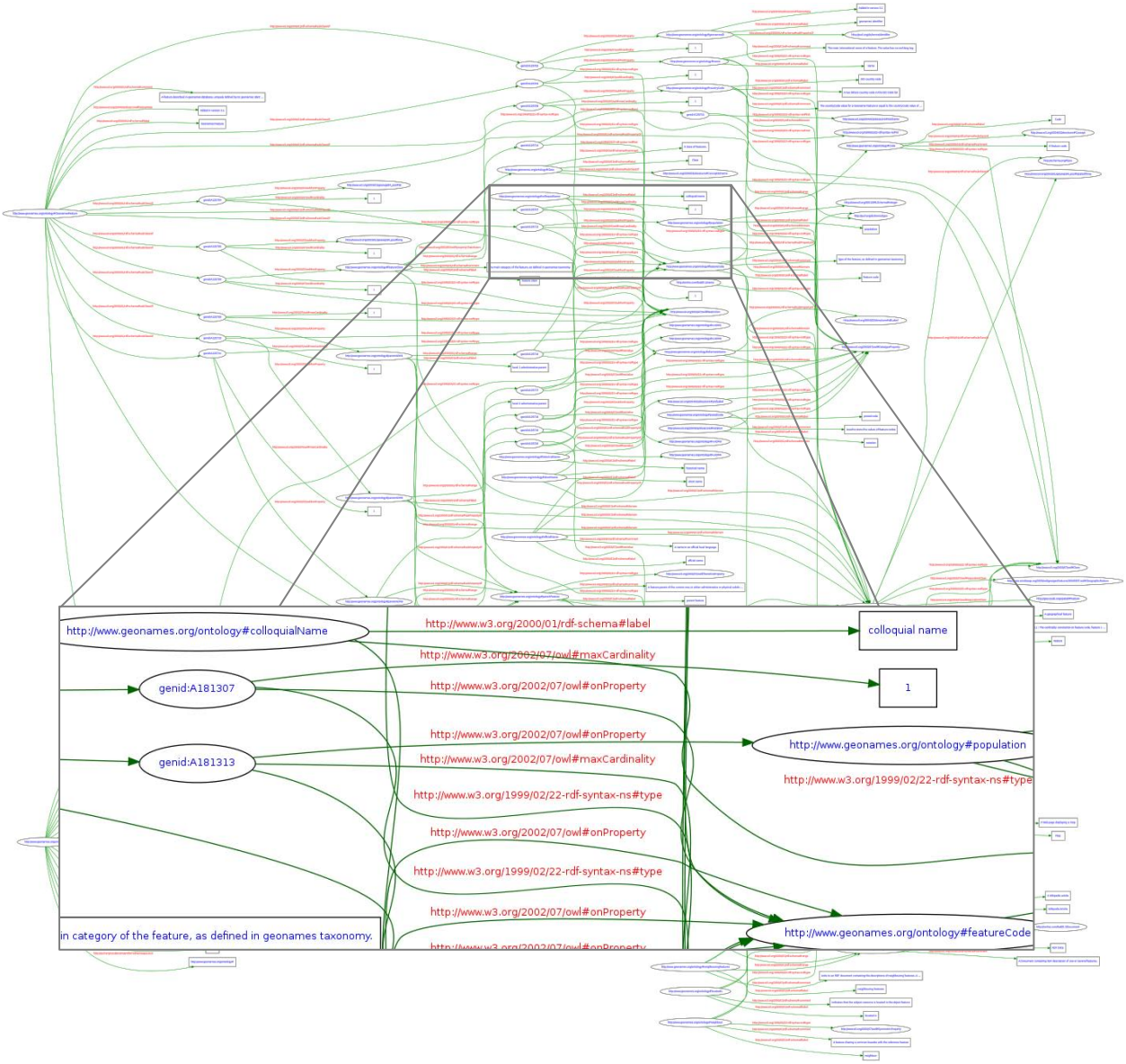


**Figure 1** – Graph structure of Classes and Attributes in the GeoNames Ontology, for illustrative purposes. Nodes represent subjects and objects while edges represent predicates. Note that nodes can be both subjects and objects.

Figure 1 is an excerpt from the GeoNames Ontology [5], a widely used ontology for describing geographic places, and illustrates the graph structure common to OWL ontologies.

OWL itself it built upon another fundamental semantic web standard; the *Resource Description Framework* (RDF). RDF is the W3C recommendation for describing and exchanging information resources using Internet protocols. It also provides a flexible method for representing knowledge by decomposing knowledge into the smallest possible components, called statements. Statements consist of subject-predicate-object sets, or *triples*, that represent simple, distinct facts. Each statement relates a subject to an object. The predicate describes the nature of the relationship between the subject and the object (e.g. the triple <Light water reactor> <isType> <Nuclear reactor> represents the fact that a light water reactor is a class of nuclear reactors.) In order to make this method functional, it is necessary that subjects, predicates and objects (each potentially defined in OWL) be uniquely identified by a *uniform resource identifier* (URI). URIs are simply text strings that identify and provide the location of resources. Depending on the intended use of the resource identified by the URI, the URI can point to a location on the Web, in a file system, or on an internal server. Often, these URIs may refer to locations of classes or individuals in an OWL ontology. Specifying entities in this way allows for knowledge to be reused, shared and distributed across resources (for example, a subject

may come from one location while an object and predicate come from another). In place of an object, predicates my relate subjects to a value. However, these values may belong to a class, as dictated by the predicate's logic.

Note that RDF is not a file format itself but is publishable in many different file formats, most commonly XML (other serializations such as N3 and Turtle also exist).

As mentioned previously, OWL is built upon RDF and allows for more expressive and meaningful descriptions and definitions of classes, properties and relationships than is provided by RDF itself. Additional specifications that are also RDF based such as the *Simple Knowledge Organization System* (SKOS) can be used within RDF documents to express additional information [6]. SKOS, for example, provides support for creating classification schema and thesaurus-type knowledge for resources.

Once ontologies have been developed and resources have been described using RDF, each can be queried using SPARQL, an RDF query language [7]. SPARQL is to RDF as SQL is to relational databases (RDF databases are referred to as *triplestores*). SPARQL, however, has many unique capabilities as compared to SQL that reflect the utility of RDF and semantic technology. For one, since RDF allows knowledge to be distributed, SPARQL can query across several repositories (i.e. it can conduct federated searches). In this way, a single query can draw information from multiple locations. Also, since semantic technology allows information to be integrated based on the meaningful descriptions encoded in RDF, SPARQL may potentially be used to generate complex queries for information from separate but interacting domains, given that each of these domains share a common ancestral schema.

Finally, there are several tools designed to reason over RDF data and OWL ontologies using the logic inherent in these specifications [8]. These reasoners compute logical consequences (e.g. if-then-therefore relations) from the statements contained within the triplestores or ontologies and allow for the development of sophisticated capabilities such as hypothesis testing or scenario generation based on the state of knowledge.

## 3 Examples of ontology-driven semantic technology adoption

Several diverse scientific disciplines and commercial industries have adopted semantic information technology in order to handle vast amounts of disparate and heterogeneous information and to create solutions to complex problems.

Perhaps the richest area of applied ontology work comes from the biomedical field. The problems of information management in clinical medicine are not entirely distinct from those of safeguards. Biomedical professionals need to integrate large amounts of heterogeneous and distributed information in order to come to high-confidence conclusions regarding a patient's treatment. Medical records come in the form of imagery, laboratory results, unstructured text, and the like, and are decentralized, likely residing in individual doctors' offices. The need to unambiguously and reliably describe, communicate and exchange complex medical terminology is the primary driver behind this work [9]. The development of biomedical ontologies affords a solution to many of these problems by providing shared vocabularies and a standardized exchange mechanism for communicating terminology in a universal, reliable, and reusable manner.

Scientists in ecology are leveraging semantic information technology to synthesize knowledge from the volumes of existing information collected by individuals and institutions around the globe [10]. Because field research is difficult, costly and generally takes place over long time periods, scientists are looking at new ways to generate knowledge beyond collecting new data. Ecoinformatics has emerged as a subfield that is largely dedicated to integrating existing data to test new hypotheses, and to building tools to manage and share information based on semantic standards, e.g. [11]. Ontologies provide this community with a standard for publishing data so that it can be easily discovered and integrated with other resources that exist in various locations and in various formats as well as a new means of conducting science.

Not surprisingly, some of the largest and most conspicuous purveyors of information are adopting semantic technologies to help handle distributed, heterogeneous information. Within the past year

Google introduced the Knowledge Graph project that seeks to organize information resources and relationships among them in order to provide richer search results to its users by generating graphs based on search queries [12]. Facebook has developed the Open Graph Protocol, a simplistic metadata format based on RDF that allows web developers to describe their resources in a standardized way so that each information resource published on the web might be integrated into the social graph of Facebook [13].

For more information on the application of semantic information technology, W3C maintains a list of use cases that showcases the diversity of application spaces [14]

## 4 Semantic safeguards

There are several areas of safeguards for which the utility of semantic technology can be clearly recognized. Open-source information is one of these areas as it requires the collection and management of large amounts of heterogeneous, largely unstructured information. Given a well-developed nuclear fuel cycle ontology, open-source information could be semantically tagged with relevant concepts by natural language processors and stored with RDF metadata, greatly increasing the ability to integrate this information with other sources. Further, such a capability could be applied to existing resources, and information could be queried and combined regardless of where the information resides (similar to a federated information system).

Semantic technology also holds great potential for handling information in various languages by using internationalization standards that are already in place across the Web. For example, as the most common serialization of RDF is XML, any property can be modified by the xml:lang attribute to add additional translations to any resource. Once such a translation is added to a concept within the ontology, each resource referencing the concept can automatically inherit this translation and these resources can be seamlessly queried using any language encoded in the ontology.

The adoption of a nuclear fuel cycle ontology is consistent with the State-Level Approach. A nuclear fuel cycle ontology representing the general knowledge of all fuel cycle elements, materials, facilities and technologies can be instantiated for each state with a safeguards agreement. State-specific knowledge (i.e. the states existing fuel cycle activities, declarations, etc.) could be added to each state-level ontology. This could serve as the platform for all state-level safeguard information to be stored, integrated and queried, thus facilitating the sharing of knowledge throughout the organization. Previously mentioned reasoning engines could then automatically analyze the consistency of declarations in light of existing knowledge, and highlight potential areas of concern for a given state (i.e. state X has technology Y and therefore may be doing Z). Such an ontology would inherently contain all possible proliferation pathways allowing for such capabilities as simulating diversion scenarios, and supporting future planning of safeguards activities by indicating to analysts and planners where to seek out potential noncompliance. This would represent an objective yet differentiating means of safeguards implementation.

Knowledge management at the enterprise scale would also be supported by the adoption of semantic information technologies. A standard, shared conceptualization of the nuclear fuel cycle, or of a particular state's fuel cycle as mentioned above, could serve as a valuable training tool for new staff. It could also be used as a tool to capture knowledge of departing staff. Moreover, semantic technology can standardize the exchange of information between departments, creating greater interoperability between systems which will be increasingly important under the State Level Concept.

## 5 A path forward

Ontology-based semantic information technology has the *potential* to impact every aspect of safeguards. While these benefits have already begun to be recognized with the use of named entity extraction and information harvesting tools, the full impact of semantic technologies may require the concerted, collaborative effort of the safeguards community.

Broadly speaking, it can be argued that the nuclear fuel cycle is the primary knowledge domain of interest for IAEA safeguards. By explicitly defining and describing existing knowledge of the nuclear fuel cycle—that is, by developing a nuclear fuel cycle ontology—the full potential of semantic information technology can begin to be reaped. However, the development of such a large and complex ontology is far from trivial. While the development of a complete fuel cycle ontology may be an ambitious undertaking it may be achievable by leverage the collective knowledge of the global community of safeguards experts, and by utilizing existing knowledge resources (such as the IAEA's Physical Model and various IAEA-maintained databases).

Ontology development must necessarily be a consensus activity. Creating an Ontology Working Group, perhaps under the auspice of the INMM-ESARDA (with input and guidance from the IAEA, of course), would provide the organizational platform and access to the expertise necessary to achieve this goal. This group could establish a collaborative engineering approach, breaking up the project into sub-tasks and sub-domains.

Noy and McGuiness [15] offer a sequential process for ontology development which includes: (1) Defining the domain and scope of the ontology; (2) Reusing existing ontologies and resources (for example, the Suggested Upper Merged Ontology [16] or some other upper-level ontology for high-level modeling); (3) Enumerating important terms in the domain; (4) Defining classes and class hierarchies; (5) Defining properties of classes (including cardinality, range, restrictions, etc.); and (6) Creating instances based on classes.

Following this model, the Working Group could make decisions regarding the first two tasks, while relying on subgroups with expertise in specific fuel cycle elements (e.g. enrichment, reactors, conversion, etc.) to carry out the third. Once the important domain terms are determined, the final steps could be "crowdsourced" to experts around the globe who could contribute knowledge in their areas of expertise. To achieve this, a wiki-type site (with access controls) could be established with templates for specifying classes, properties and relationships. If done correctly, these wiki-pages could be easily converted into RDF/OWL documents. Such an effort would need to be curated to control quality, with editors volunteering for specified time periods to review additions and changes.

A parallel effort could involve IT specialists developing an information architecture capable of utilizing the ontologies generated by the group. This need not involve a complete re-engineering of the established architecture as semantic technologies may extend current systems rather than replace them.

## 6 Conclusions

Berners-Lee stated that "standards are the basis of emerging technologies" [17]. Developing a nuclear fuel cycle ontology—a standard, shared knowledge representation of the nuclear fuel cycle—may be the key driver of innovation toward addressing the information management challenges of safeguards and making the safeguards regime significantly more efficient. While this is a difficult task, by leveraging the existing knowledge resources and the collective intelligence and effort of the global safeguards community, this can be achieved.

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# Development of an Advanced Ceramic Seal for Maintaining Continuity of Knowledge in Treaty Verification and Safeguards Applications

**Heidi A. Smartt, Juan A. Romero, Maikael Thomas, Billy Cunningham**

Sandia National Laboratories
P.O. Box 5800 MS1371
Albuquerque, New Mexico 87185 USA
E-mail: hasmart@sandia.gov

***Abstract:***

*Sandia National Laboratories and the Savannah River National Laboratory are collaborating on research and development of technologies for an advanced capability prototype tamper-indicating device known as the Ceramic Seal. Advanced capabilities include multiple levels of tamper indication such as a frangible seal body, surface coatings, and an active detection of seal status; and unique identification via electronics as well as non-reproducible surface features. The innovation of the Ceramic Seal is the inclusion of multiple advanced capabilities in a volume comparable to the ubiquitous metal cup seal. Our advanced capability small volume seal has application in treaty verification and safeguards regimes for maintaining continuity of knowledge. Once attached to a monitored item, the seal's identity and status can be verified in-situ rather than requiring removal and analysis at an inspectorate location.*

*The Ceramic Seal has evolved from a first generation prototype constructed of alumina to a second generation prototype manufactured from low-temperature co-fired ceramic (LTCC). LTCC allows integration of passive electronic components into the seal construction material. Vulnerability reviews have been conducted periodically throughout the project and results used to guide the design. This paper will describe the capabilities of the current generation Ceramic Seal.*

**Keywords:** Tamper-indicating devices; seals; Containment and Surveillance

## 1. Introduction

Containment/surveillance (C/S) measures aim to ensure Continuity of Knowledge (CoK) or Chain of Custody (CoC) during inspector absence on the movement of nuclear or non-nuclear material, weapons throughout their lifecycle, equipment and samples, and preserving the integrity of relevant data. Viewed as complementary to nuclear material accountancy, C/S is a critical element of many non-proliferation regimes. C/S equipment and approaches require continuous improvements because (1) the adversary continues to technically advance (which could render C/S equipment obsolete with a single technical advancement), (2) requirements could change based on the introduction of new procedures or approaches, and (3) as technology advances there may be new options for C/S equipment, including options that provide efficiency gains.

The Ceramic Seal [1,2], a collaborative effort between Sandia National Laboratories (SNL) and the Savannah River National Laboratory (SRNL), integrates multiple advanced technologies into a prototype next generation loop seal with various technical options available depending on the deployment. One possible deployment is as a replacement for the ubiquitous metal cup seal (Figure 1) used by various organizations. The metal cup seal, although environmentally robust, inexpensive, and small in size, is operationally burdensome and its integrity is not able to be verified in-situ. The Ceramic Seal addresses issues with the metal cup seal and makes additional security advancements (tamper indication and unique identification) and efficiency improvements (in-situ verification and ease

of application). Its innovation is the integration of these advanced capabilities in a small volume, including a self-securing wire feature; multiple levels of tamper indication via a frangible seal body, surface coatings, and active detection of state through low power electronics; electronic identification number verified in-situ through a contact reader, and physical identification via non-reproducible surface features. This paper will discuss the status of the research and development of the various technologies and integrated prototype loop seal.

**Figure 1: Size comparison of a metal cup seal (left), rapid prototype of an early model of the Ceramic Seal (right), and a U.S. quarter (bottom).**

## 2. Advanced security

The most critical element of a seal applied in a treaty verification regime is its tamper-indicating features. A loop seal will employ a wire or fiber-optic cable (FOC) threaded through a monitored item's hasp or otherwise secured, and the wire or FOC will terminate within the seal body. In single use seals such as the Ceramic Seal and metal cup seal (versus multiple use seals in which the seal wire can be removed and reattached), confidence must be maintained that the wire is unable to be removed from the seal body once secured without detection and that the seal body has remained intact such that the seal body has not been opened and the wire removed/replaced. Tamper-indicating features on the seal body serve the role of providing this confidence. The following subsections describe the advanced tamper-indicating features in use by the Ceramic Seal.

### 2.1. Seal body

The body of the Ceramic Seal is fabricated using a low-temperature co-fired ceramic (LTCC) process. With LTCC, passive components such as resistors, capacitors and inductors along with conductive line traces can be integrated into a monolithic package, thereby freeing valuable space for active components and the battery. Furthermore, LTCC packaging can reduce electronic noise and radiated emissions from the device [3].

The properties of the material used in LTCC meet the requirements of "frangibility" – that is, upon deformation it tends to break into fragments rather than retaining cohesion, yet the material is strong enough to withstand the operational environment. Frangibility is important so that a tamper attempt might result in difficult-to-reassemble fragments. Other materials also meet these requirements; however, they do not have the added benefits of LTCC, namely the ability to integrate electronics. The first prototype of the Ceramic Seal was constructed of 99.8% $Al_2O_3$. The current version, using DuPont 951 LTCC Green Tape™ material, is manufactured using, by weight, 50.3% $Al_2O_3$, 31.3% $SiO_2$, 11.9% PbO, 1.1% $K_2O$, and 5.4% CaO [3].

The LTCC manufacturing process involves punching unfired or green ceramic sheets with via holes, screening with a conductor, stacking with excellent precision, laminating the result into a monolithic three-dimensional structure, and firing in a furnace. Figure 2, Figure 3, and Figure 4 show the Ceramic Seal during the LTCC process.

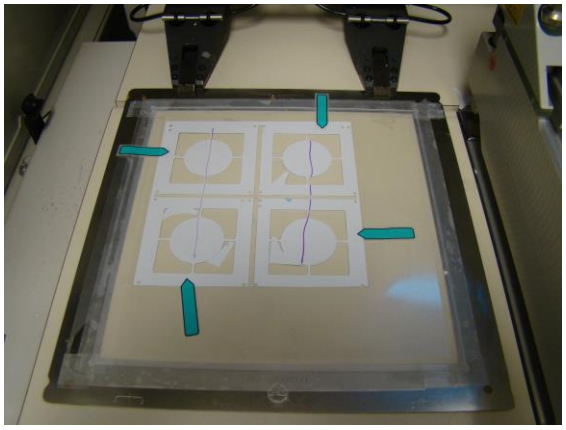

**Figure 2: LTCC Green Tape™ punching process.**

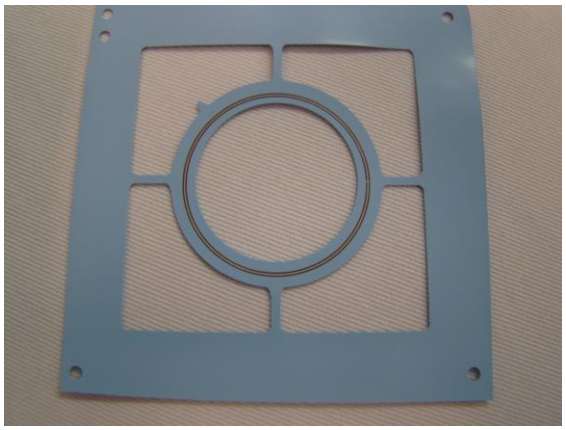

**Figure 3: Conductive trace screen printing.**

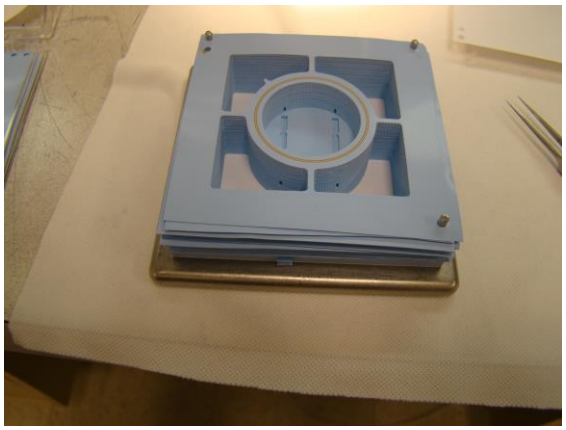

**Figure 4: Green Tape™ stacking prior to lamination.**

## 2.2. Tamper planes

As mentioned above, conductive line traces can be integrated into the LTCC – some are used to connect electronic components and a separate set are embedded throughout the body of the seal for tamper indication (tamper planes). These tamper planes are connected to the electronics and if disrupted, i.e., signals cannot pass, software within the electronics impacts performance.

## 2.3. Active tamper indication

The Ceramic Seal provides active tamper indication. Physically, the electronic components are attached to the embedded line traces in the seal cap (see Figure 6 for electronics housing). The Ceramic Seal utilizes a single microcontroller. Two battery springs make contact with a battery directly below the electronics.

Seal firmware is programmed prior to deployment; however, the Ceramic Seal requires “personality programming” in-situ, meaning configuration must happen via the RS232 serial communication vias located on the cap of the seal. Personality programming loads the secret keys onto the seal, sets message creation interval, and sets absolute time. The electronics will not be powered until the seal cap and base is connected, so personality programming the seal must happen after it has been closed. However, for added security, we have designed the seal to accept personality programming only one time, i.e. cryptographic keys can only be loaded a single time.

The seal creates several message types – state-of-health (configurable), anomalous events, and the seal interrogation history. As messages are created, we append a message authentication code (MAC) using the 128-bit CMAC algorithm with AES cipher (and optionally encrypted using 128-bit AES) before storing in flash memory. The MAC derives its uniqueness from the secret key, the seal’s 8 byte ID, a non-repeating message count and a clock. The 8 byte ID is assigned during firmware programming and can be a unique number by procedure. The MAC ensures that the seal itself can be uniquely identified due to the combination of the 8 byte ID with the cryptographic key.

A seal reader, which will also have the secret keys, will be able to send an authenticated command to the seal (over the serial port), receive the requested message(s), and authenticate them using its copy of the secret key.

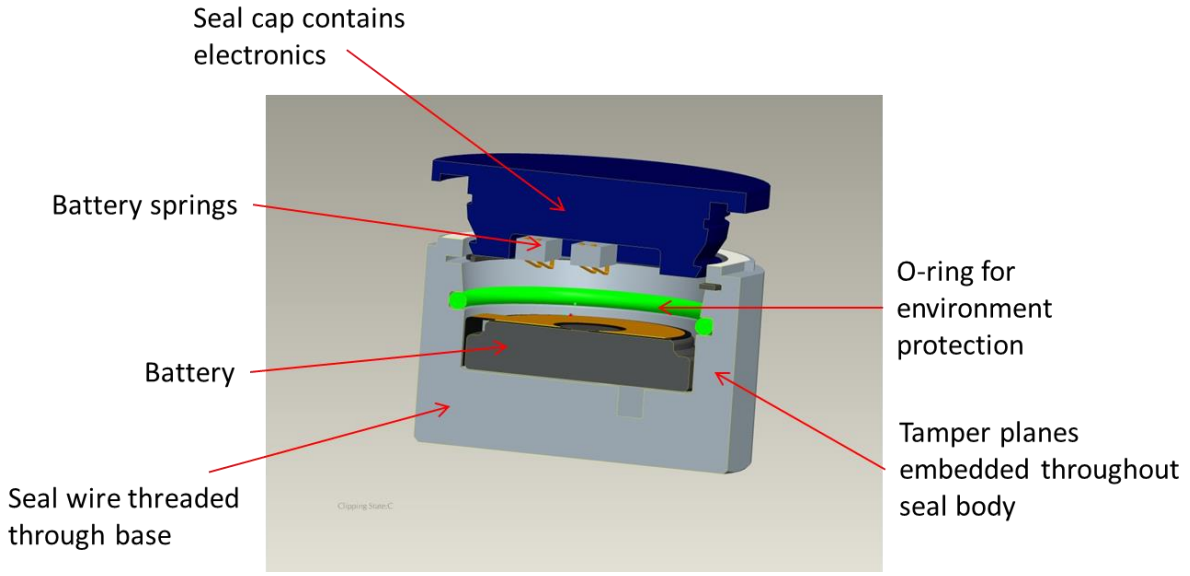


**Figure 5: Seal concept. Image courtesy SRNL.**

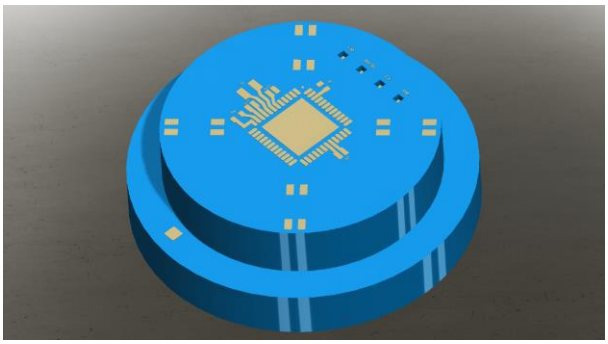

**Figure 6: Seal cap, fabricated using LTCC, with electronics underside.**

### 2.4. Coatings

SRNL is developing coatings [4,5] for application to the outer surface of the seal body. Modification to the seal body will be noticeable under a fluorescent light. As the coatings are in development, they will not be discussed in this paper.

## 3. Improved efficiency: self-securing wire

The capability of self-securing wire not only improves efficiency but touches upon security as well. The wire ends must securely terminate in the seal body in such a manner that they cannot be easily removed, and must do so in an efficient manner. In the Ceramic Seal design, the wire is routed through the monitored item and into the seal base, where it is secured by a tortuous path. The design team and SNL vulnerability review (VR) team iterated on several designs before choosing the method shown in Figure 7. The wire itself is important as well. Active research in identifying appropriate wires is on-going and commercial candidates have been identified.

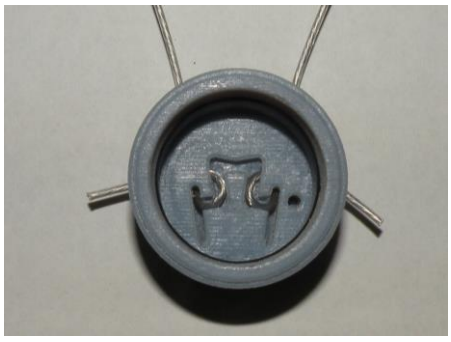

**Figure 7: Rapid prototype of seal base highlighting self-securing wire mechanism. Image courtesy SRNL.**

## 4. Improved efficiency: in-situ verification

The Ceramic Seal allows in-situ verification with a contact reader to verify its integrity. Currently, the reader is implemented using a serial connection and a laptop computer. As described in section 2.3, the seal reader will contain the same secret keys as the seal. It will be able to send an authenticated command to the seal over the serial port, receive the requested message(s), and authenticate them using its copy of the secret key. The reader supports the following commands: request latest sensor state of health, request a specific message number from the seal, request that the seal send the latest anomalous/tamper message, request that the seal send all anomalous/tamper messages stored on the seal, and personality programming.

## 5. Next steps

The next step in the design is to complete development of the Ceramic Seal software. Once complete, the VR team will review the electronics and software and provide guidance for any modifications, if any, to the electronic and software design team.

The seal prototype will be fabricated and assembled using LTCC and will consist of tamper planes, the electronics, battery, and coatings from SRNL. The completed prototype will undergo functional testing, and then a comprehensive VR. Final modifications will be made to the seal based on guidance from the VR team.

## 6. Acknowledgements

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Unless otherwise noted, all pictures or images have been taken or created by Sandia National Laboratories.

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# Development of a Non-contact Reader System for Reflective Particle Tags

**Heidi A. Smartt, Michael Sinclair, William Sweatt, Keith Tolk, Juan Romero, Karl Horak**

Sandia National Laboratories
P.O. Box 5800 MS1371
Albuquerque, New Mexico 87185 USA
E-mail: hasmart@sandia.gov

***Abstract:***

*Reflective particle tag systems can employ thousands of microscopic, randomly located and oriented reflective elements suspended in a matrix to create a unique pattern, to be read and verified with an optical system. Sandia National Laboratories developed the original Reflective Particle Tag (RPT) in the 1990's to tag treaty-limited items under the START framework. The RPT has evolved with advances in computing technology, imaging, and material science, and in its current generation is considered a robust, low-cost, high security passive tag for treaty verification and safeguards applications.*

*The current RPT reader system docks to the frame encompassing the tag. In some situations, however, a non-contact reader system is preferred. Sandia National Laboratories is currently researching such a non-contact reader system for tags based on reflective elements and developing a prototype system. Interrogation of the tag will include illumination with multiple wavelengths and multiple incidence angles. The new approach expands the information collected from tags beyond spatial locations and orientations of the particles, to include spectral and morphological attributes. To address image registration challenges, a low concentration of larger, fluorescent "tracer particles" are added to the tag to allow for precise alignment. This paper discusses the research and status of the prototype system.*

**Keywords:** Reflective particle tag; Containment and Surveillance

## 1. Introduction

Containment/Surveillance (C/S) measures are critical elements of any monitoring regime to ensure Continuity of Knowledge (CoK) during inspector absence on verifying movement of nuclear material, weapons, equipment and samples, and preserving integrity of treaty-relevant data. Their continual improvement is required because (1) the adversary continues to technically advance (which could render C/S equipment obsolete with a single technical advancement), (2) requirements could change based on the introduction of new procedures or approaches, and (3) as technology advances there may be new options for C/S measures, including options that provide efficiency gains or allow deployment in new application spaces.

A tag is one such C/S measure and is intended to establish identity of an item as accountable, maintain CoK of status of that item over time, and provide evidence of tampering. Tags may provide evidence of tampering with the item if applied in such a manner, e.g. across a seam of a container, and must provide evidence of tampering with the tag itself, e.g. counterfeit and duplication.

The Reflective Particle Tag (RPT) was developed in the 1990's to identify (tag) treaty-limited items under the START framework. The tag has proven resistant to duplication, counterfeiting, and removal without detection. It is stable through temperature extremes, rough handling, and years of use. As a

passive tag, it requires no energy source while attached, making it a unique and robust option for many inspection applications.

The RPT is a field-applied tag composed of specular hematite particles in a clear, adhesive polymer matrix (Figure 1). A reader (based on a custom camera and illuminators) is physically attached to the tag frame for alignment and illuminates the tag from four angles (Figure 2). When illuminated by the reader, each tag presents complex patterns of millimetre-scale light reflections unique to the tag. These patterns are used to physically authenticate the tag. The reader acquires and transfers a set of 4 images per tag (one image per illumination angle). A unique identifier (ID) at the midline identifies the tag and enables automated tag reading. An inspector can subsequently return to the item, attach the reader, compare IDs, then reflective patterns to determine if tag sets match (verify or reject).

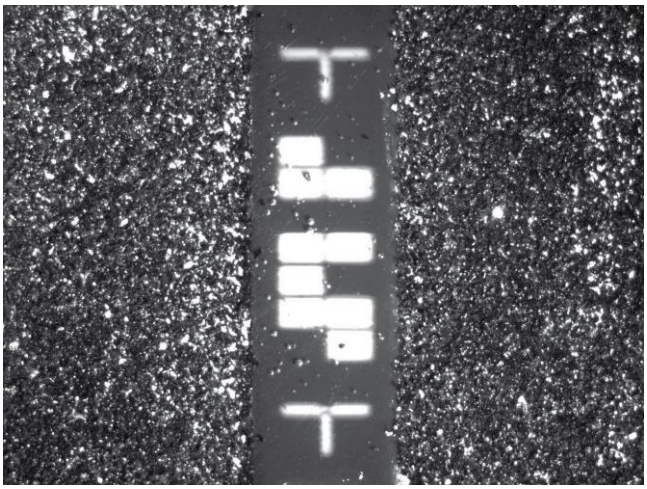

**Figure 1: Reflective particles create unique patterns that are difficult to duplicate. A strip located in the middle of the tag contains a unique binary code ID.**

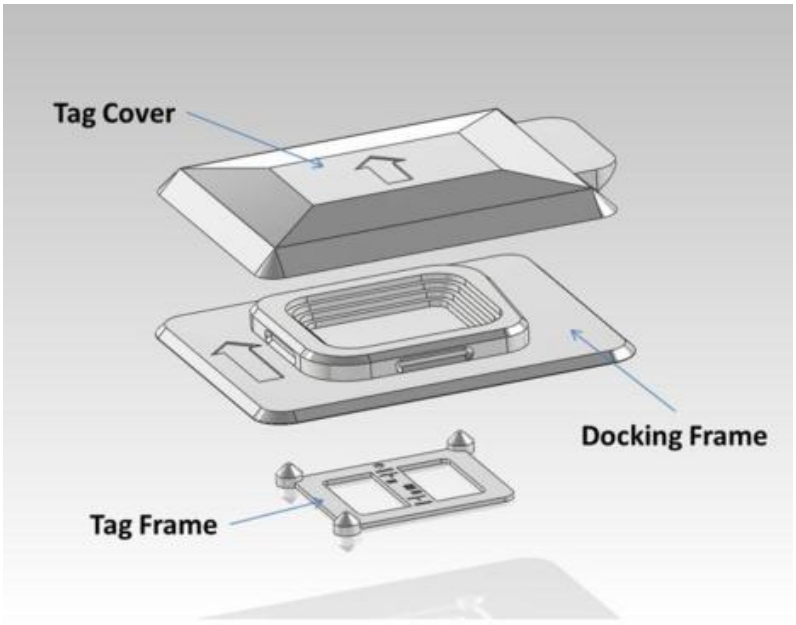


**Figure 2: RPT tag cover, docking frame, and tag frame.**

The combination of a counterfeit-resistant reflective tag with an integrated unique ID in a passive, robust technology makes RPT the appropriate choice for applications with strict facility acceptance requirements and for deployments in which a semi-permanent tag should be attached to an item's

surface. However, it is embodied in a frame which limits its use in some complex geometric applications, e.g. curved surfaces. Furthermore, verification of RPT requires physical contact with the RPT docking frame attached to the tagged item. The RPT system can be enhanced by (1) offering the option of removing the frame, which allows broader deployment options such as application to curved surfaces (Figure 3), and (2) developing a non-contact reader system. In some treaty verification scenarios, physical contact of the reader to the item may not be desired nor allowed. A non-contact readout system can minimize the time that inspectors spend in harsh or environmentally restricted locations, and allows automation, e.g. reading seals on $UF_6$ cylinders. In some scenarios, the host applies the seal and reads it in view of an inspector and does not want the inspector touching the monitored item. Furthermore, in some scenarios, certification becomes easier if the reader does not touch the monitored item.

**Figure 3: "30B" type $UF_6$ cylinder. Tag would typically be placed on the end cap for protection. Image courtesy Westerman Companies.**

The purpose of this project is to develop a system with a tag that can be applied to complex geometries, read with a non-contact readout system, and meets additional verification regime requirements such as security, durability, low cost, and ease of application.

## 2. Challenge 1 – Removing Frame

There are two primary challenges associated with removal of the frame – (1) the tag is not confined to application on a flat surface and thus might be curved or uneven, and (2) the reader must know how to initially orient itself relative to the tag for acquiring the reference image and subsequently how to re-orient itself in the same position for verification. Furthermore, as the reader is handheld and non-contact, images must be acquired rapidly as the user is limited in ability to maintain stability above the tag.

We will address the first issue by acquiring multiple images with varying parameters at each location. Acquisition of multiple images at each location also partly addresses the second challenge since it helps to compensate for tip/tilt errors once the reader is in place. Another method for addressing the second challenge is to redesign the illumination beams. We have a candidate design that is currently being fabricated for testing.

Finally, we can use fiducials for alignment in all directions – both translation and rotation. We are considering two types of options – either embedding a small number of large fluorescing particles into the matrix, or more standard fixed fiducials such as those used in the current RPT system (see Figure 1). The larger fluorescent particles will be randomly mixed into the tag. The tag will be illuminated using blue light from a GaN (gallium nitride) LED with wavelength on the order of 400nm. The resulting fiducials will fluoresce in the visible and will be for accurate positioning of the reader and finding focus.

If we proceed with the standard fiducials, we can use Augmented Reality (AR), COTS processing tools, and tools developed for the RPT system for registration. A prototype AR tool has been developed to aid in manual alignment of our optical system. The AR system uses the RPT fiducial as a frame of reference for projecting a 3D image within a camera's view screen. When the images are in the correct location in the view screen, the camera is at the correct x, y, z coordinates and oriented properly in roll, pitch, and yaw. The user only has to move the camera to the proper position using

real-time optical feedback in order to obtain the proper location and orientation of the unit for repeatable exposures. We will next need to interface the developed application with our actual system such that when the user is “locked” into the correct position our system will trigger and the camera will begin imaging. As a further benefit, using the RPT midline fiducial allows us to borrow the RPT algorithm for acquiring tag ID number.

The final issue is user stability while positioned above the tag. Based on our initial design, we believe that we can acquire a sequence of exposures in less than 1 second. The user should be able to hold the camera reasonably steady for this length of time, and then the alignment algorithms can do the precise centering required for comparing the new images against the stored images taken previously.

## 2. Challenge 2 – Illumination of the Seal

The RPT reader docks directly onto a frame surrounding the tag, eliminating ambient light. For our system, we are challenged by the ambient light from undesired directions interfering with the illuminators. We approach the issue by replacing the white LED illuminators used in the RPT reader with multiple bright LEDs (the response of red, green, and blue are shown in Figure 4). A candidate illumination system has been designed and fabricated.

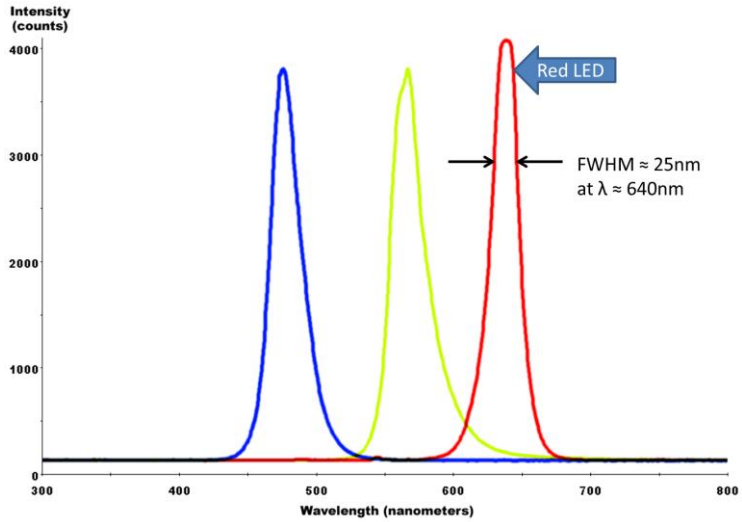


**Figure 4: Emission spectra of commercial red, green, and blue LEDs. Image from Wikimedia.**

The multiple LEDs in each illuminator location are physically separated, but our initially-designed compact optical system will allow the illumination patterns from the LEDs to overlap nicely. Without this correction, the images acquired from each LED would not co-register and the reflective particles that “light up” will be slightly different for each LED image. This has been a difficult challenge to overcome.

## 3. Challenge 3 – Maintaining Security

Due to the redesign of the illumination and optical system, the handheld readout system will be more tolerant to tilts, rotations, and translation. Hence, the spatial pattern of reflections recorded by the detector will remain relatively stable over a small range of angles. In contrast, the current RPT system will yield a completely different spatial pattern even with minimal movement. Thus, the current RPT system is capable of discerning smaller orientation errors of particles within a potentially counterfeited tag. In order to maintain security, we chose to explore options besides the position and reflection of the particles, including spectral properties, shape of the particles, and topographical information. This approach requires examination of particle type, camera resolution, and exploitation of already-available information.

We are exploring two types of reflective particles: (1) hematite particles used for the RPT system and (2) engineered spectrally selective faceted particles. The use of these spectrally selective particles will allow us to investigate the value of including channels of spectral information in the recorded images. Although we have designed and fabricated a set of these particles, they have not yet been tested and will not be discussed further in this manuscript. Instead, we will focus on the existing micaceous specular hematite.

Micaceous hematite particles of approximately 80 micron size were used for initial testing. Hematite is an attractive particle as it is highly reflective; its shape is non-uniform and varies widely with particle; and it exhibits flat facets for strong specular reflections. For increased security we propose to exploit the variability of the particle shapes, and record spatial information regarding each particle's shape. Using this approach, the comparison of new and reference images will also include the shapes of each of the particles. To achieve this, the readout optical system employs a high resolution camera.

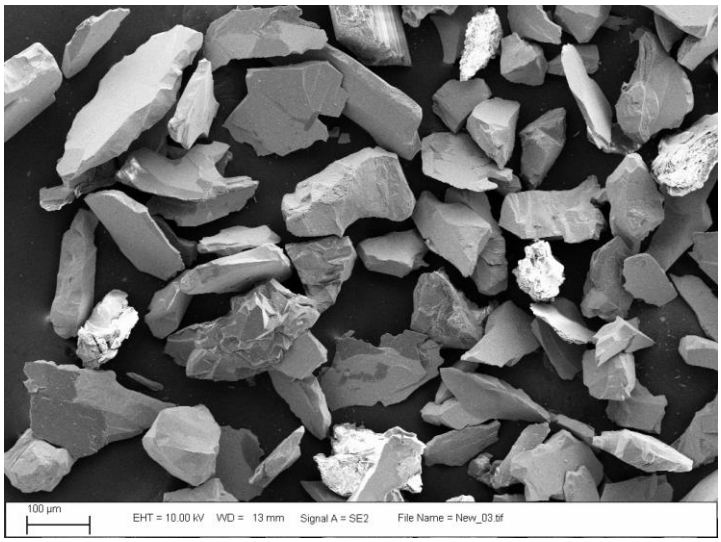


**Figure 5: Specular hematite.**

## 4. Next Steps

The next steps in this project can be divided into six primary activities: (1) benchtop optical system assembly and testing, (2) testing of engineered spectrally selective faceted particles, (3) fluorescent particle development for fiducials, (4) continue development of registration tools, (5) conceptualize miniaturization, and (6) begin development of interface architecture for software.

## 5. Acknowledgements

Unless otherwise noted, all pictures or images have been taken or created by Sandia National Laboratories.

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# Testing a direct method for evaluating the concentration of boron in a fuel pool using scintillation detectors, and a $^{252}$Cf and an $^{241}$Am-Be source

**Dina Chernikova[1], Kåre Axell[1,2], Imre Pázsit[1], Anders Nordlund[1], Petty Cartemo[1]**

[1]Chalmers University of Technology, Department of Applied Physics, Nuclear Engineering, Fysikgarden 4, SE-412 96 Gothenburg, Sweden
[2]Swedish Radiation Safety Authority, SE-171 16 Stockholm, Sweden
E-mail: dina@nephy.chalmers.se

***Abstract:***

*The present investigations are aimed at the development and testing of a direct non-destructive method for evaluating the concentration of boron in a fuel pool using scintillation detectors. The method uses a modified ratio between two gamma lines with energy of 480 keV and 2.23 MeV. These lines belong to the capture of a thermal neutron in boron and hydrogen, respectively. The relation between them can reveal the concentration of boron in the fuel pond.*

*The method proposed was tested in a laboratory experiment with a $^{252}$Cf and an $^{241}$Am-Be source. EJ-309 liquid scintillation detectors were used for measurements of gamma spectra. The concentration of boron in water varied from 1550 ppm to 4200 ppm. The optimization and test studies were performed via MCNPX simulations.*

*The results of these tests are provided in the present paper and they show that the boron content in water can be determined through using the characteristics of gamma lines with energy of 480 keV and 2.23 MeV.*

**Keywords:** boron; liquid scintillators; EJ-309; PSD

## 1. Introduction

There exist a number of non-destructive assay techniques for evaluation of irradiated and non-irradiated nuclear fuel. Some of them, such as the Fork Detector P. Rinard [1], the Under Water Neutron Coincidence Counter (UWCC) G. Eccleston [2] and the Cherenkov viewing device E. Attas [3], may be used directly on the site of a nuclear fuel pool.

However, working under the conditions of the fuel pond is related to some difficulties, e.g. the intense radiation field and the presence of a neutron absorber, usually boric acid. Boron affects several processes: multiplication, detection efficiency, die-away time and the doubles/totals ratio in the case of UWCC technology (UWCC based on the detection of thermal neutrons by $^{3}$He counters) I. Koulikov [4]. This influence is normally not significant for the results while measuring with fast scintillation detectors. Though, in order to be able to make necessary corrections of the results it is still important to evaluate independently the boron concentration in the water of the nuclear fuel pond.

In a recent paper we described the basic principles of the concept suggested for the evaluation of the boric acid concentration in water using two gamma-lines in the gamma spectrum obtained by a scintillation detector D. Chernikova [5]

## 2. The main concept

The main idea of the proposed method consists of using two thermal neutron capture gamma-lines with energies 480 keV and 2.23 MeV in

the gamma spectrum obtained by the scintillation detector.

The 2.23 MeV gamma line characterizes the capture of thermal neutrons in hydrogen. It has two sources, one of them is the scintillation detector itself. Scintillation detectors contain a large amount of hydrogen. Therefore, part of the high-energy neutrons coming from the source (nuclear material, nuclear fuel, radioisotope source etc.) get slowed down to near thermal energies mainly by elastic scattering in the detector itself. At thermal energies, the neutrons diffuse through the material until they undergo thermal capture. When a hydrogen atom captures a thermal neutron, it turns into deuterium with the release of 2.23 MeV gamma-rays which is finally detected by the scintillator, as shown in Fig.1.

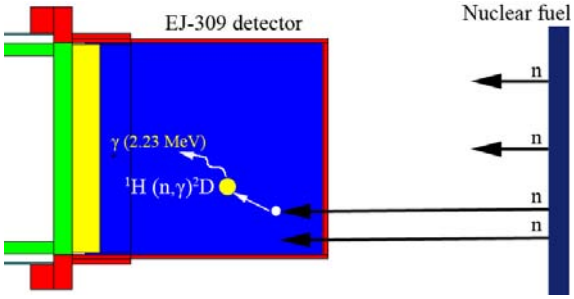


**Figure 1:** a simplified illustration of the characteristics of the 2.23 MeV gamma line for proposed concept (source #1).

If there is pure water between the nuclear material and the detector, then part of the neutrons arising from the source get slowed down and then captured by hydrogen atoms already in water, with a release of the large fraction of 2.23 MeV gamma-rays, as shown in Fig.2. It corresponds to the second and the main source of gamma-rays with energy 2.23 MeV.

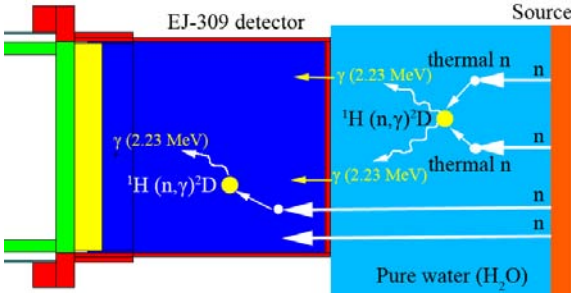


**Figure 2:** a simplified illustration of the characteristics of the 2.23 MeV gamma line for proposed concept (source #2).

However, if the water contains boric acid, some of the thermalized neutrons will be captured by boron atoms in the reaction $^{10}B(n,\alpha)^7Li$. After absorption, about 94% of the reactions leave the $^7$Li ion in its first excited state, which in $10^{-13}$ seconds de-excites to the ground state by releasing a 480 keV gamma ray, as shown in Fig. 3.

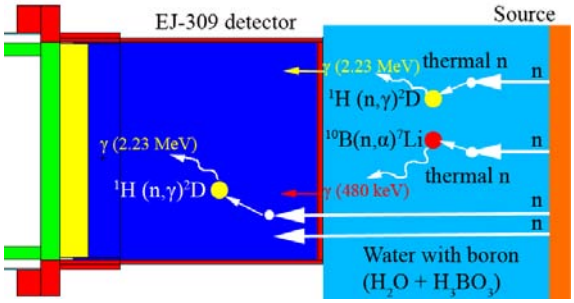


**Figure 3:** a simplified illustration of the characteristics of the 480 keV gamma line for the proposed concept.

Eventually, the capture of thermal neutrons will be shared between boron and hydrogen with a boron capture being favoured due to high cross section. The more boron that is dissolved in the water the more thermal neutrons will be captured by boron atoms with the release of 480 keV gamma rays and less of them will undergo thermal capture in hydrogen. Thus, the ratio between these two gamma lines might be used to reveal the concentration of boron in water.

## 3. Experimental setup

The concept described above was originally tested in experiments with an $^{241}$Am-Be (α,n)-source with the neutron yield of approximately $1.1 \cdot 10^7$ neutrons per second D. Chernikova [5]. The source was enclosed in a lead capsule 11 cm high with a diameter of 7 cm and placed approximately 18 cm below the top side of a polyethylene container of height 42.5 cm and width 43 cm, as shown in Fig. 4.

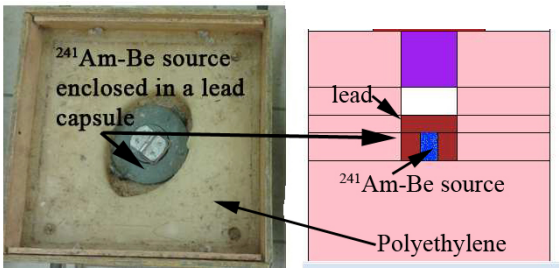


**Figure 4:** geometry of the $^{241}$Am-Be (α,n)-source used in D. Chernikova [5].

In this work the experiment was repeated with a $^{252}$Cf spontaneous fission source with a neutron

yield of approximately $9 \cdot 10^6$ neutrons per second, which is shown in Fig. 5.

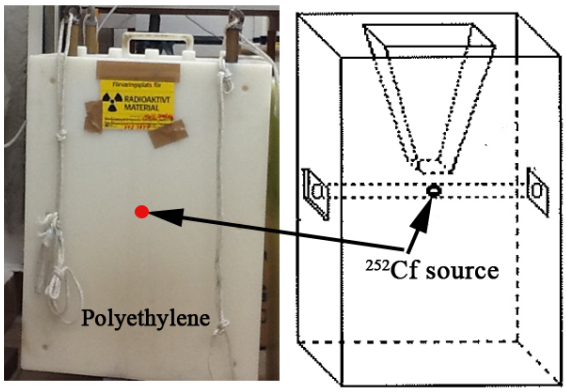


**Figure 5:** geometry of the $^{252}$Cf source used in the present work.

In both cases the same experimental set-up was used, i.e. the detector was placed at a distance of about 25 cm from the top of the container. Between the source and the detector, an Al tank of 25 cm height was installed. The Al tank was filled with 4.2 liter of pure or borated water, as shown in Fig. 6.

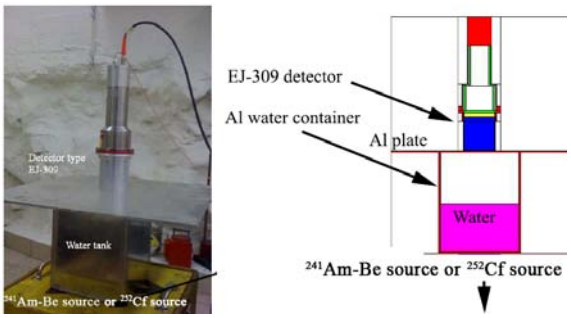


**Figure 6:** photo and schematic drawing of the configuration of the experimental and simulation set-up (for both, the $^{241}$Am-Be and the $^{252}$Cf sources).

The concentration of the boron content in water varied from 0 to 4200 ppm.

In experimental studies with the $^{252}$Cf spontaneous fission source we used the same electronics, settings, procedure and EJ-309 liquid scintillation detectors, as have been used in experiments with the $^{241}$Am-Be (α,n)-source D. Chernikova [5].

## 4. Results and discussion

In the experiments a number of data sets were obtained for different concentrations of boron in water. Each data set includes the charge calculated within the long gate, the charge calculated within the short gate and Pulse Shape Discrimination (PSD) index.

The data obtained in experiments with the $^{241}$Am-Be source showed that the use of various ranges of PSD index can significantly change the results of evaluation. As an example, in the current study for both sources, the $^{241}$Am-Be and the $^{252}$Cf, the data with PSD index between 0.044 and 0.096 allow better evaluation of boron concentration.

The explanation of this observation can be related to the fact that the PSD index is tied to a relation between the delayed and prompt fluorescence. Therefore, further on it is important to make a sensitivity and optimization study in order to evaluate a reliability of use a PSD index as a “tuning parameter”.

Fig. 7 shows the PSD profile obtained for the $^{252}$Cf source when the Al tank is filled with pure water. An analogous profile was built for experimental data obtained while using the $^{241}$Am-Be source.

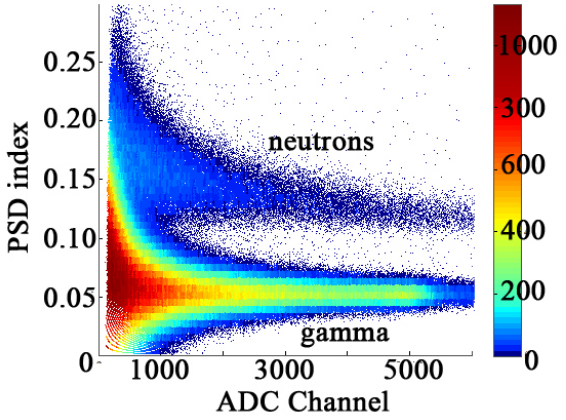


**Figure 7:** 2D plot of the counts as a function of the pulse energy (ADC channel) and PSD parameter for the $^{252}$Cf source.

From these profiles only gamma rays having the PSD parameter in a specific range were selected. For experimental studies performed with the $^{252}$Cf source, the PSD parameter varied from 0.04 to 0.07, while for cases when the $^{241}$Am-Be source was used the range of PSD indices was between 0.044 to 0.096.

Due to this procedure, a substantial part of the low energy gammas were eliminated from the spectrum, as one can see in Fig. 8 obtained for measurements performed with the $^{252}$Cf source.

However, the methodology described above was sufficient to test the method.

As a final step of evaluation, the background subtraction procedure was developed based on the characteristics of gamma lines and

assuming that the shape of the background curve for each particular concentration of boron in water will be proportional to the analogous value obtained for the water case.

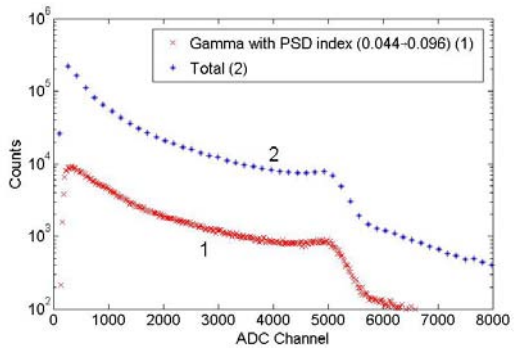


**Figure 8:** Total and only gamma spectrum with PSD index which varied from 0.044 to 0.096, obtained by the EJ-309 detectors for the case when the Al tank was filled with pure water (experiment).

The procedure was slightly different for the two sets of measurements with the $^{241}$Am-Be and the $^{252}$Cf sources due to the presence of a 4.44 MeV gamma line in the spectrum of the $^{241}$Am-Be source. Therefore, for the data obtained with the $^{241}$Am-Be source, first, the Compton edge of the 4.44 MeV gamma was removed using the best fit linear equation.

In all other aspects the background subtraction procedure was identical for both sources. That is the peak at 2.23 MeV for the water case was normalized to the same peak but obtained for a specific concentration of boron in water, as shown in Fig. 9. Finally, the Compton scattering spectrum of the 480 keV gamma line was obtained by the subtraction of the normalized water spectrum from the actual spectra.

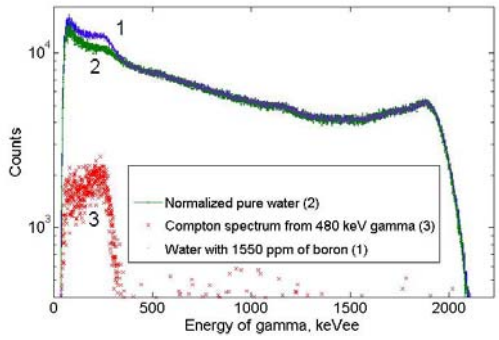


**Figure 9:** gamma spectrum for 1550 ppm of boron in water after background subtraction.

As a final result of evaluation, the ratio of the two gamma lines was obtained for a few different cases, as shown in Fig. 10. The blue line in Fig. 10 corresponds to the expected boron concentration in water (input data) normalized to 1550 ppm. The rest are the ratios of the measured (or simulated) count rates which have been normalized to the value of the count rate obtained for the case of 1550 ppm of boron dissolved in water.

For the $^{241}$Am-Be source, three different data sets are presented in Fig. 10. The violet curve corresponds to the MCNPX simulations D. Pelowitz [6] which were performed for the $^{241}$Am−Be source in real geometry D. Chernikova [5]. Due to the lack of capability to simulate the whole scintillation process with inclusion of delay and prompt fluorescence, it was not possible to calculate the PSD index for each detected gamma ray.

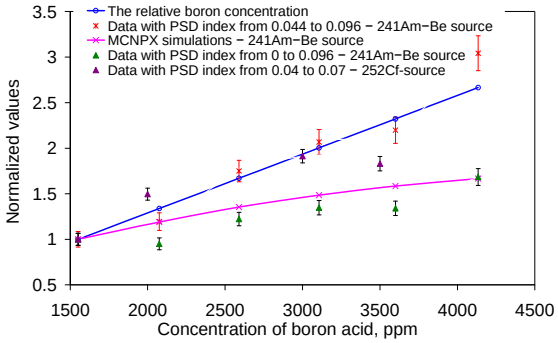


**Figure 10:** normalized value of the intensity of gamma rays with energy 480 keV to the intensity of gamma rays with energy 2.23 MeV and the relative boron concentration in water normalized to 1550 ppm (real change of boron content) in water (experiment and simulations).

Thus, for purpose of comparison with experimental results the violet curve can be compared to the measurement data with PSD index from 0 to 0.096 (green triangles) obtained with the use of the $^{241}$Am-Be source. These data show that the concept proposed is capable to predict the concentration of boron between 1550 and 4000 ppm, though the results (green triangles and the violet curve) is far from the expected ("ideal") boron concentration in water normalized to 1550 ppm (blue curve).

However, if the same data are evaluated within the different range of PSD index, then final curve (red dots) tends towards the "ideal" (blue curve).

The same situation is observed for the results obtained when using the $^{252}$Cf source. However, the overall results are much worse compared to the one obtained when using the $^{241}$Am−Be source. This could be explained by the fact that the $^{252}$Cf source capsule placed at the distance of more than 26 cm farther from

the detector (due to radiation protection reasons) compared to the $^{241}$Am-Be source capsule. This agrees with the results of MCNPX simulations performed in order to optimize the distance between the $^{252}$Cf source and the scintillation detector. The geometry of the set-up used in the simulations is shown in Fig 11.

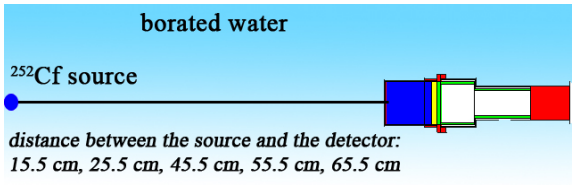


**Figure11:** schematic drawing of the configuration of the simulation set-up.

In the MCNPX simulations the $^{252}$Cf source was modelled to be a point isotropic source of both neutrons and gamma rays. In the simulations the ENDF/B-VII data libraries have been used. The results of the simulations are normalized to one source neutron. The simulations of the spectrum of gamma rays entering the detector (the tally F4) were performed for five cases, when the distance between the source and the detector varied from 15.5 cm to 65.5 cm. Distances below 15 cm were not considered due to the safety restrictions when using nuclear fuel as a source.

The results of a simulation shown in Fig. 12 indicate that the intensities of the gamma lines with energy of 480 keV and 2.23 MeV vary with distance between source and detector. It is related to the amount of source neutrons being thermalized.

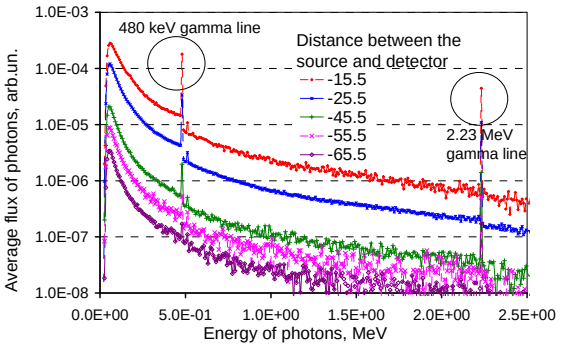


**Figure 12:** the gamma spectrum entering the detector.

Thus, as one can see in Fig. 13, for the same amount of boron dissolved in water the ratio between gamma lines with energy of 480 keV and 2.23 MeV changes from 4 to ~1 when the distance between source and detector varies from 15.5 cm to 65.5 cm. At the same time, it is shown in Fig. 13, that the source almost does not affect the ratio when the distance between source and detector approaches 50 cm.

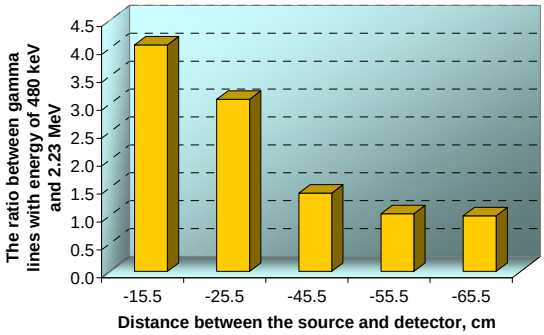


**Figure 13:** the ratio between gamma lines with energy of 480 keV and 2.23 MeV at various distances between source and detector.

Therefore, the optimal distance between the source and the detector should be equal to 15 – 25 cm.

## 5. Conclusions

In the present study we have tested a direct non-destructive method for evaluating the concentration of boron in a fuel pool using scintillation detectors and a $^{252}$Cf and an $^{241}$Am-Be sourcs.

The results of these tests showed that the characteristics of gamma lines with energy of 480 keV and 2.23 MeV might be used to evaluate the boron content in water of a fuel pond within a few percents. However, the use of various ranges of PSD index can significantly change the results of evaluation. As an example, in the current study for both sources, the $^{241}$Am-Be and the $^{252}$Cf, the data with PSD index between 0.044 and 0.096 allows better results of evaluation of the boron concentration.

Overall results of evaluation appeared to be much worse for the $^{252}$Cf source compared to the one obtained while using the $^{241}$Am−Be source. That required an additional optimization study to be done in order to determine the optimal distance between the source and the detector. As a result, it was shown that for the same amount of boron dissolved in water the ratio between gamma lines with energy of 480 keV and 2.23 MeV changes from 4 to ~1 when the distance between the source and the detector varies from 15.5 cm to 65.5 cm. Thus, optimally, in order to get accurate results of evaluation the source should be placed at the distances between 15 cm to 25 cm.

It is also noted that additional studies are needed in order to evaluate a reliability of the use of various ranges of PSD index.

## Acknowledgement

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# A laboratory device for developing analysis tools and methods for gamma emission tomography of nuclear fuel

**Peter Jansson, Staffan Jacobsson Svärd, Sophie Grape, Ane Håkansson**

Div. of Applied Nuclear Physics, Dept. of Physics and Astronomy, Uppsala University
Box 516, SE-75120 Uppsala, Sweden
E-mail: peter.jansson@physics.uu.se

## Abstract

Tomography is a measurement technique that images the inner parts of objects using only external measurement. It is widely used within the field of medicine, and may become important also for nuclear fuel verification where inspectors can obtain information from fuel assemblies' inner sections without dismantling them.

At Uppsala University, Sweden, a laboratory device has been built for investigating the tomographic measurement techniques on nuclear fuel. The device is composed of machinery to position model fuel rods, activated with Cs-137, in a fuel assembly pattern according to the user's choice. The gamma radiation from the model fuel assembly is collimated to a set of detectors that record the radiation intensity in various positions around the fuel model. Reconstruction of the gamma activity distribution within the fuel model is performed off-line.

The objective for constructing the laboratory device was to support the development of tomographic techniques for nuclear fuel diagnostics as well as for nuclear safeguards purposes. The device allows for evaluating the performance of different data-acquisition setups, measurement schemes and reconstruction algorithms, since the activity content of each fuel rod is well known.

For safeguards purposes, the device is unique in its capability to model various fuel geometries and configurations of partial defects. The latter includes removed, empty and substituted fuel rods. It is well suited for developing tomographic techniques that are optimized for partial defect detection. It also allows for development of analysis tools necessary to quantify detection limits.

Here, we describe the capabilities of the laboratory device and elaborate on how the device may be used to support the nuclear safeguards community with the development of unattended gamma emission tomography.

**Keywords:** safeguards; gamma emission tomography; nuclear fuel; partial defect; tomographic algorithms

# 1. Introduction

## 1.1 Introduction to tomography on nuclear fuel assemblies

Tomography is a well-established measurement technique, suitable for obtaining sectional images of the inner structure of items without the need to open them up or with other means performing destructive analysis. The technique is widely used in the field of medicine but also in other areas such as sensor technology, electronics, materials science etc.

Gamma emission tomography on nuclear fuel assemblies has previously been proven useful for both validation of core simulators and for verification of the assembly integrity within the safeguards regime [1,2]. For both purposes, the analyses are based on the recording of gamma radiation from fission fragments such as $^{137}$Cs, $^{154}$Eu or $^{140}$Ba/$^{140}$La inside the fuel rods. Using advanced calculation algorithms for tomographic reconstruction, it is possible to obtain detailed results on the rod-wise distribution of selected gamma-ray emitting isotopes, and since the abundance of these elements depends directly on the irradiation in the reactor, information on properties such as pin-wise power or burnup distribution can be obtained. Furthermore, cross-sectional images can be produced, displaying the source distribution and thus the fuel rod configuration. These capabilities of the technique are important both from the safeguards perspective and from the operator perspective; the safeguards community needs to verify the absence of undeclared activities involving the fuel assembly, and the operator community can use the data to validate simulation codes. Recently, tomography on complete fuel assemblies has also been suggested as a means for measuring fission-gas release non-destructively [3] and for identifying failed fuel rods by a reduction in their fission gas content [4].

In contrast to the here-discussed measurements on complete fuel assemblies, tomography of individual nuclear fuel rods have been performed [5] in order to validate simulation codes used for prediction of in-core fuel-rod behavior. Such measurements can provide detailed information on individual fuel rod level, but may be prohibited by safety constraints and high cost. Accordingly, for some investigations, it may be advantageous to be able to perform tomographic studies on complete fuel assemblies without the need for dismantling the assembly.

The ability of the technique to deliver information on a single pin level using non-destructive assay of the complete assembly may become important, e.g. in the verification process of spent nuclear fuel before it is sent to geological storages or other difficult-to-access storages, if the authorities require such information. However, as mentioned, a tomographic measurement device detects radiation from fission fragments and not the fissile content itself. Accordingly, it could be argued that a possible diverter could dismantle the fuel in order to extract the fissile material and replace the diverted material with fission products in addition to e.g. fresh or depleted uranium, a scenario that might not be identified using tomographic imaging. On the other hand, such a procedure is expected to be extensive and complex, in addition it would be difficult to re-create a realistic distribution of fission products in the assembly. If such a scenario is considered a serious threat, a means for its detection is to request operator-declared data on the initial content of the assemblies and on the operation of the plant, thus verifying also that the activity distribution within the fuel is in agreement with declarations.

## 1.2 Motives for constructing a laboratory device for tomographic measurements

Several motives for constructing a laboratory device for tomographic measurement were identified, leading to the manufacturing of such a device at Uppsala University:

- To allow abundant measurements on various fuel assembly geometries.
- To test new algorithms for tomographic reconstruction.
- To optimize tomographic reconstruction algorithms and measurement schemes to allow for highest possible accuracy in rod-activity measurements.
- To investigate the system's response to various configurations of possibly removed and/or replaced rods, which can advantageously be modeled in a laboratory fuel mock-up,
- To test design options for reducing the size and complexity level of tomographic equipment which is an important aspect of equipment to be used in a non-stationary configuration [5].

- To offer possibilities to simulate collective as well as individual deviations of the fuel rods from their nominal positions and to study the effect this will have on the tomographic reconstruction results.

The device and its performance are described in this paper.

## 2. Design and construction of the device

One of the most important design goals was that results from measurements using the laboratory device should be transferable to a measurement situation on actual nuclear fuel assemblies. This means that efforts were made to as closely as possible model the real measurement situation. In this case, the primary goal of the laboratory device was to investigate the capabilities of measuring the power distribution, and accordingly, materials and gamma-ray sources have been selected for this purpose. However, for demonstration and further development of the tomographic technique, these selections are somewhat arbitrary.

### 2.1 General design requirements

The requirements that were identified for the laboratory device are accounted for in table 1.

| | |
|---|---|
| Control and steering of the measurement device | Automation system without the need for human intervention |
| Storage of fuel rods between measurements | In a radiation shielded environment |
| Gamma attenuation | As realistic as possible, i.e. depicting attenuation in a real measurement situation |
| Lateral positioning precision of detectors | 0.1 mm |
| Angular positioning precision of assembly relative to detectors | 0.1° |

**Table 1:** Design requirements for the laboratory device.

The first two requirements in table 1 arise from the need to offer adequate radiation protection for the personnel working with the device. The third requirement had significant implications on the selection of activity and materials of the model rods, which is further discussed in section 2.3 below. The last two requirements were set in order to allow for calibration and for studies of the repeatability of the technique. However, one should note in this context that the readout of each position could be made with even higher precision. This level of positioning precision is in accordance with what was obtained for a tomographic measurement device that was used for measurements on commercial fuel assemblies, as reported in [7].

### 2.2 Mechanical design and hardware

The tomographic laboratory equipment is constructed using stainless steel components, and has a robust system of girders to support the high accuracy movements that are necessary for the tomographic measurements. Firstly, there are three axes for picking up individual model rods from a lead-shielded storage and placing them in the desired configuration on a magnetic table (and replacing them in the storage after the completion of a measurement). Secondly, there are two axes for performing the tomographic measurements; one rotational axis for the magnetic table onto which the model fuel rods are positioned during measurements, and one translational axis with the collimator-detector system for collecting gamma-ray intensity projections at each selected rotation angle about the fuel model. The setup is illustrated in figure 1. The lid of the rod storage is pneumatically controlled, as is the grip that holds rods during movement from the storage to the measurement position.


**Figure 1:** Images of the device. 1) Storage. 2) Table. 3a-3c) Linear axes for moving rods. 4) Linear axis for translation. 5) Axis for rotation. 6) Collimators. 7) Detectors. The right image is a zoom of the magnetic table and the collimators and detectors. The middle image shows the grip placing some model rods on the magnetic table.

## 2.3 Model rods

As mentioned in the introduction, one of the purposes for tomographic measurements on nuclear fuel assemblies is to verify core simulators on a fuel rod level. An established method for validation of the calculated power distribution is to measure the distribution of $^{140}$Ba/La, which is strongly related to the distribution of (heat) power [8]. This is done by measuring its 1596 keV gamma-ray emission line, which has previously been demonstrated using tomographic techniques [9].

Because of difficulties related to the manufacturing of model fuel consisting of actual nuclear fuel materials (uranium dioxide surrounded by a cladding of zircaloy) and loading it with a gamma-ray source of the relevant energy, it was decided to instead work with rods that contain $^{137}$Cs. The gamma decay energy from this isotope is 662 keV i.e. significantly lower than for $^{140}$Ba/La, and in order to adapt to that, a rod filling material was selected that mimics the attenuation coefficient of 1596 keV gamma radiation in the authentic materials. Here, a bronze granulate was selected for modeling the fuel material and a Titanium alloy for modeling the cladding material. These selections result in gamma-ray attenuation coefficients that are presented and compared to authentic coefficients in table 2 below.

| | Authentic fuel | | Fuel model | |
|---|---|---|---|---|
| Activity | $^{140}$Ba/La (1596 keV) | | $^{137}$Cs (662 keV) | |
| | Fuel | Cladding | “Fuel” | Cladding |
| Material | UOX | Zircaloy | Bronze granulate | Titanium |
| Attenuation coefficient [cm$^{-1}$] | 0.560 | 0.298 | 0.381 | 0.342 |
| Mass density [g/cm3] | 10.47 | 6.53 | 5.22 | 4.54 |

**Table 2:** Materials and attenuation coefficients for $^{140}$Ba/La gamma rays in authentic fuel, respectively for $^{137}$Cs gamma rays in the laboratory device. Attenuations coefficients are from reference [10].

The rods were manufactured by pouring a water solution containing a well-known activity of $^{137}$Cs into a container with the amount of bronze granulates that was to be filled in a batch of model rods. After mixing and drying, the bronze granulate, now with the activity of $^{137}$Cs homogenously distributed was filled into the rods, which were then sealed with a top plug. In total, 70 model rods were manufactured in about ten batches, with activities ranging from 54% to 126% of the average activity.

A set of measurements was performed to study the degree of $^{137}$Cs activity homogeneity within the rods. A selection of rods was scanned axially with a horizontal collimator in front of a HPGe detector. The results showed that the distribution of $^{137}$Cs was homogeneously distributed, as demonstrated in figure 2.

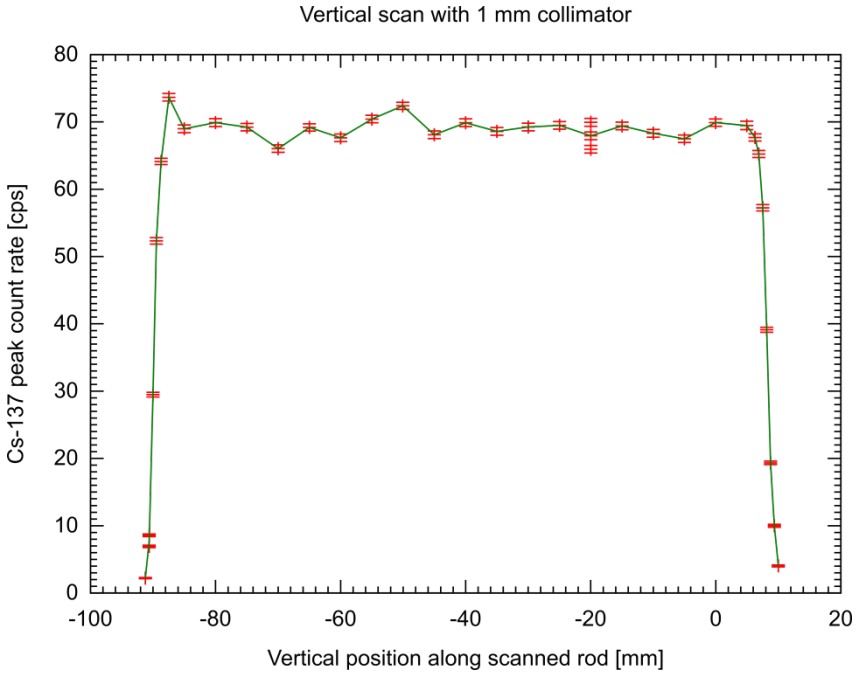


**Figure 2:** The experimental results (red marks) of an axial scan of a model rod. The scan was made axially using a HPGe detector, defining the view of the model rod using a collimator slit with a height of 1 mm.

After manufacturing, the fuel rods were measured individually in order to obtain reference activity values for the tomographic measurements, firstly using a HPGe detector in a separate setup and secondly using BGO detectors in the tomographic setup. At each occasion, the counting rates and thus the dead times of the data collection systems were low. At each campaign, one rod was measured repeatedly throughout the campaign, resulting in a repeatability of 0.4% and 0.2%, respectively (1 σ). The agreement between the rod activities in the two data sets was 0.9% (1 σ). Some systematic uncertainties that may cause this discrepancy are a non-linear response to counting rate in either of the setups, different sensitivities to the individual heights of the stack of bronze granulates in the rods, changes in the compactness of the bronze granulates in between the two campaigns and/or imprecision in positioning in either of the campaigns. Still, the reference activities were considered accurate enough to offer an experimental benchmark of the tomographic measurement technique. Here, the HPGe data has been used as the reference, mainly due to that the positioning system in that campaign was considered to be more reliable.

## 2.4 Collimator and detector design

The detector package is constructed using steel with a density of about 8 g/cm$^3$ and is equipped with a collimator with four slits and four corresponding scintillator detectors as shown in figure 1. Each slit has a different width (0.1, 1, 2 and 3 mm, respectively) in order to study the influence of the collimator width on the tomographic measurement results. In addition, extra collimator modules are available, allowing for the future use of identical slit widths for the detectors.

In the investigations presented here, BGO (bismuth germanium oxide, $Bi_4Ge_3O_{12}$) scintillator detectors were used. Considering the current source of $^{137}$Cs emitting a single gamma-ray energy at

about 662 keV, these detectors were deemed sufficient in spite of their relatively poor energy resolution.

### 2.5 Automation and control system

The automation system, schematically displayed in figure 3, is responsible for the movement of the axes, for controlling the rotating magnetic table and for the pneumatic systems that maneuvers the rod grip function and the opening/closing of the lead storage cask.

The power output modules used to give current to the servo motors contain a PID regulator to place the axes in selected positions. Communication with the power output modules is performed via a serial interface (RS-232) to each module [11]. On each axis, optical switches can be used to locate a reference position that is used for subsequent movement of the axis, including the rotating magnetic table.

### 2.6 Data acquisition

As described above, the radioactive substance in the model rods is $^{137}$Cs, which has a single peak in the gamma-ray energy spectrum at about 662 keV. Due to the simplicity of the emission spectrum, a simple data acquisition system based on EG&G ORTEC 850 (Quad SCA) single channel analyzers was selected to count pulses from the detectors. However, in this context it should be noted that the data-acquisition system is easily exchangeable, allowing for analyses of the capabilities of the technique when using different systems. Figure 3 includes a block diagram of the current data acquisition system. The number of pulses analyzed by the SCA's is counted by a PC-TIO-10 card from National Instruments Inc. in the control computer.

In a future upgrade of the device, we plan to use multi channel analyzer cards, e.g. from [12], which, compared to using SCA's, would allow for more specific analysis of the 662-keV peak. In addition, such hardware would allow for energy-dependent analysis of measurement results such as studies of gamma-ray scattering effects on the uncertainty of the results from the tomographic reconstruction.

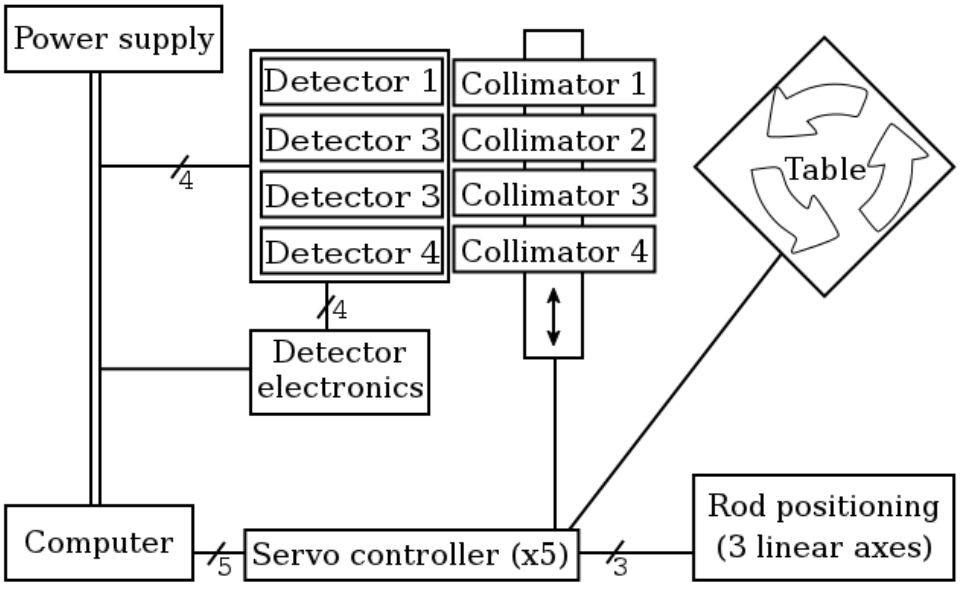


**Figure 3:** A block diagram of the data acquisition and automation system in the tomographic device.

## 3. Performance demonstrations

A set of measurements has been performed in order to demonstrate the capabilities of the device regarding its data collection characteristics, its mechanical properties and its usefulness for investigating tomographic methodology in terms of e.g. its applicability for nuclear safeguards. Some of these measurements are presented in the following sections.

### 3.1 Demonstration of data collection

As described above, the laboratory device in its current setup includes four BGO detectors, and the data collection is performed using an SCA and a counting card, implying that only the number of events collected above a certain energy threshold is stored. Thus, no spectra are recorded, although this is planned for in a future upgrade of the system. Initially, however, a separate system for recording of gamma-ray spectra was used, and the SCA output was applied to gate the spectrum collection in order to enable the selection of a proper energy threshold. The energy spectrum is presented in figure 4.

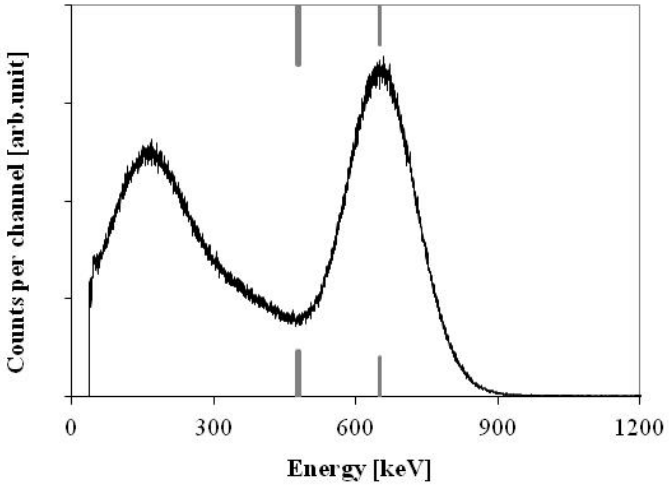


**Figure 4:** An energy spectrum collected of the $^{137}$Cs radiation from an assembly of model rods using one of the four BGO detectors in the laboratory device. The spectrum was collected for illustrative purposes, whereas the normal data collection was performed using an SCA and a counting card, thus only registering the number of events above a selected energy threshold. Two different energy thresholds used are illustrated (see section 3.5).

The energy threshold was typically set at the minimum at the low-energy side of the 662-keV peak, but in some measurements, the threshold was varied to study its effect on the results obtained in tomographic measurements. This is further described in section 3.5 below.

### 3.2 Stability with time and temperature

A study was performed of the stability of the data-collection system, including the effect of temperature on the number of counts recorded, by placing a stationary rod in front of one collimator slit and measuring continuously using the corresponding detector for 24 hours. The results are presented in figure 5.

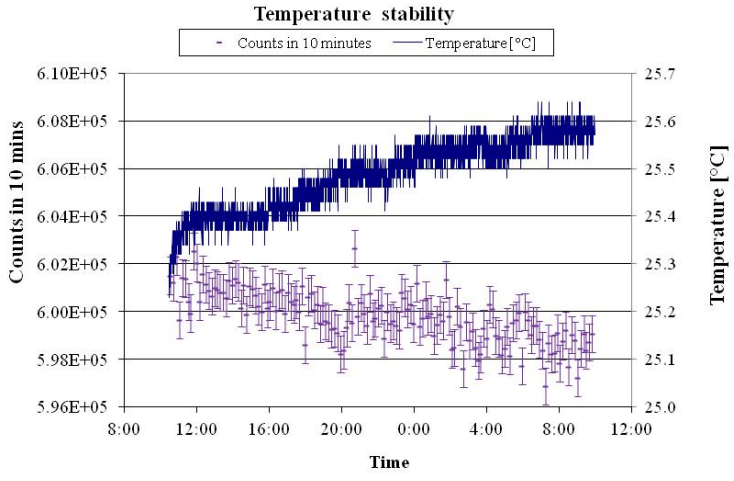


**Figure 5:** The number of counts recorded in 10-minute intervals during a continuous 24-hour measurement, illustrated together with the registered temperature during the same time period.

A temperature dependence of about 2%/°C was noted. The temperature change during a tomographic measurement, as presented below, was typically 0.5 °C, thus affecting the measurements to a relatively small extent. In the results below no corrections for this drift was made, although such corrections are possible.

### 3.3 Precision in positioning of model fuel rods

The positioning capabilities of the model rods are crucial for enabling accurate modeling of the gamma-ray transport in a tomographic measurement. Accordingly, studies were performed of the precision obtained when positioning individual model rods on the magnetic table.

Here, the position obtained for each rod was deduced by analyzing the intensity profile recorded when translating the collimator laterally in front of the rod. It was estimated that this procedure could give the position with about 0.02 mm precision. However, repeating the positioning of one individual model rod gave a repeatability of 0.08 mm and 0.04 mm in the x- and y-directions, respectively, indicating that the positioning of the rod is somewhat more imprecise. Furthermore, the individual rods showed considerably larger offsets from the intended position on the magnetic table, as presented in figure 6.

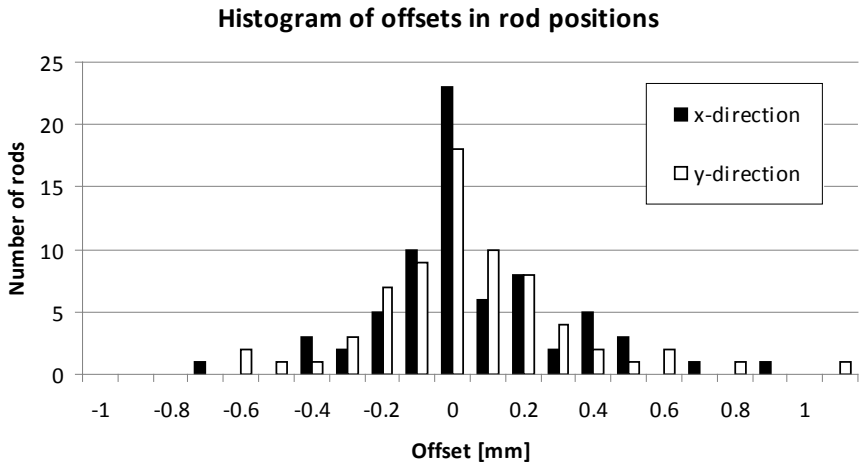


**Figure 6:** The offsets obtained in positioning of individual fuel rods. The repeatability in positioning of an individual fuel rod was <0.1 mm, indicating that imprecision in manufacturing is the likely reason for these offsets.

In a more detailed study of each rod, it was found that certain rods gave extreme offsets in both the x- and y-directions. As a consequence, the seven rods exhibiting the largest positioning offsets were excluded from the tomographic measurements presented below, implying that no rods used had an offset larger than 0.5 mm. Furthermore, the individual offsets of the rods used were taken into account when modeling the assembly in the tomographic reconstructions (see section 3.5).

### 3.4 Verifying modeling capabilities

The laboratory device allows for tests of modeling capabilities with respect to simulations of gamma-ray intensity projections. An example of a measured intensity projection is presented in figure 7 together with simulated data obtained using deterministic modeling with a point-kernel code. The simulated intensities have been scaled to the same average as the measured intensities.

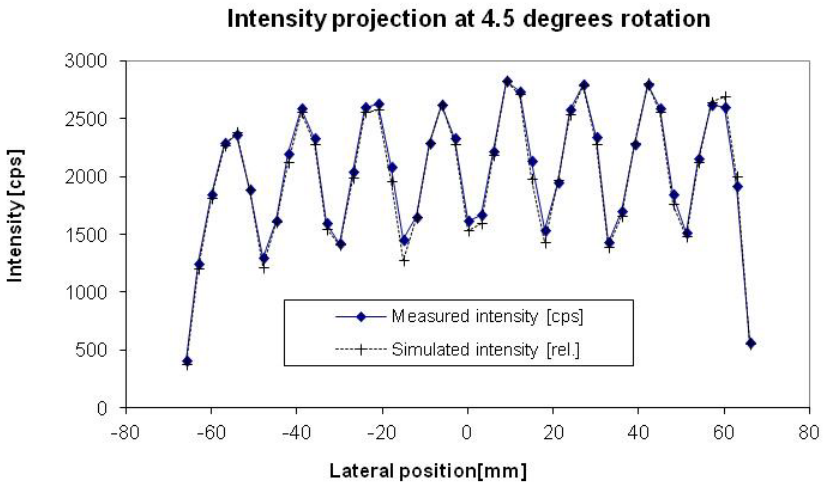


**Figure 7:** A measured intensity projection, illustrated together with the corresponding simulated projection.

As illustrated in figure 7, simulations were found to be in good agreement with the measurements.

### 3.5 Demonstration of tomographic rod-activity reconstruction

One of the main purposes with the laboratory device was to develop tomographic techniques for accurate measurement of rod-activity distributions and to investigate the achievable accuracy in such measurements. In the line of this work, algebraic algorithms were developed, allowing for rod-activity reconstructions to be made [1].

The algorithms model the full-energy gamma-ray transport of selected emission energies, and require a data-collection system with which the full-energy events can be recorded with high selectivity. However, the data-collection system applied here implies that a significant scattering component may be included; in particular if a lower discriminator level is used. (See the spectrum in figure 4.) Two modes of adapting to this were applied; (1) a high discriminator level was applied (see the upper indicated level in figure 4), or (2) effective attenuation coefficients lower than the theoretical values (table 2) were applied.

An example of the results obtained in a tomographic reconstruction is presented in figure 8. In this measurement, only one of the four BGO detectors was used, which was connected to a collimator slit of 2 mm width. In total, 2072 detector positions were used, divided upon 40 projection angles (using a 9-degree step) and 45-57 lateral positions per projection (depending on the projected width of the mockup assembly). The lateral step was 3 mm. The data collection time in each position was 20 s, giving a maximum of about 66 000 counts in any detector position.

Measured rod configuration
[relative rod activities]

| | 8 | 7 | 6 | 5 | 4 | 3 | 2 | 1 |
|---|---|---|---|---|---|---|---|---|
| H | 0.751 | 0.771 | 0.760 | 0.881 | 0.906 | 0.924 | 0.894 | 0.904 |
| G | 0.759 | 0.764 | 0.879 | 0.871 | 0.873 | 0.914 | 0.919 | 0.997 |
| F | 0.767 | 0.892 | 0.884 | 0.976 | 0.989 | 1.029 | 1.024 | 1.028 |
| E | 0.870 | 0.865 | 0.981 | 0.993 | 1.001 | 1.010 | 1.014 | 1.010 |
| D | 0.881 | 0.989 | 0.990 | 0.979 | | 0.996 | 1.106 | 1.122 |
| C | 1.013 | 0.993 | 1.002 | 1.123 | 1.139 | 1.139 | 1.096 | 1.103 |
| B | 1.015 | 1.001 | 1.126 | 1.147 | 1.127 | 1.239 | 1.200 | 1.135 |
| A | 1.001 | 1.110 | 1.151 | 1.146 | 1.207 | 1.240 | 1.221 | 1.156 |

Reconstructed rod activities
[deviation from “true” activities, %]

| | 8 | 7 | 6 | 5 | 4 | 3 | 2 | 1 |
|---|---|---|---|---|---|---|---|---|
| H | 2.1 | 0.7 | 2.0 | 1.8 | 0.8 | -1.2 | 0.9 | -0.1 |
| G | 2.5 | 1.3 | 0.2 | 1.3 | 0.8 | 0.8 | -1.1 | 0.5 |
| F | 1.3 | -1.6 | 0.3 | 0.5 | -0.3 | -1.2 | -0.1 | -0.1 |
| E | 2.6 | 1.2 | -0.3 | 0.2 | 1.6 | 0.8 | 0.2 | 0.4 |
| D | 1.0 | -0.7 | 0.7 | 2.7 | | 1.2 | -1.2 | 0.4 |
| C | -0.7 | 0.3 | -0.1 | -1.4 | -0.7 | -0.1 | 0.8 | 0.4 |
| B | -2.3 | 1.0 | -1.1 | -1.8 | -1.1 | -3.3 | -0.6 | 0.5 |
| A | -0.2 | 0.6 | -0.5 | -1.4 | -0.7 | -1.7 | -1.6 | -0.8 |

**Figure 8:** A modeled configuration of rod activities (left) and the agreement obtained between the modeled activities and the tomographically measured activities (right). All activities were scaled to an average value of one. The standard deviation obtained was 1.2%.

Furthermore, the same tomographic measurements were repeated five times on the same fuel assembly model with the purpose to study the repeatability. As a result, it was concluded that the relative activity of each rod was repeated with an average standard deviation of 0.2%, and the largest standard deviation obtained for any rod was 0.4%.

Considering the exquisite repeatability result, it is expected that the accuracy of the technique may be enhanced further as compared to the results in figure 8. Two identified ways to improve the data quality would be to employ detectors with higher energy resolution, such as $LaBr_3$ detectors, and to apply spectroscopic peak analysis, including subtraction of background under the peak.

### 3.6 Demonstration of tomographic partial-defect detection

Two tests were performed to assess the capabilities of the tomographic method for partial-defect detection, i.e. to investigate whether it can be used to detect the diversion of fuel rods. Here, the same configuration of fuel rods as presented in figure 8 was measured, i.e. the ”true” rod activities ranged from 25% below average to 24% above average, with a relative standard deviation of 13%. However, the rod in position (E,5) was replaced with (i) an empty mockup rod, and (ii) a non-active mockup rod (i.e. filled with non-active bronze granulate).

Just as above, gamma-ray intensities were recorded in 2072 detector positions, but here, the measurement time in each position was 10 s, implying a maximum number of counts in any position of slightly more than 30 000.

One may foresee that images of the gamma-ray source distribution may be useful to a safeguards inspector, and accordingly, image reconstruction was performed of the measured data. Some images obtained with the rod in position (E,5) replaced are presented in figure 9, together with an image obtained with an intact fuel assembly model, using the data recorded as presented in section 3.5, above. Here, six gray-scale levels have been applied, covering the whole range of reconstructed activities.

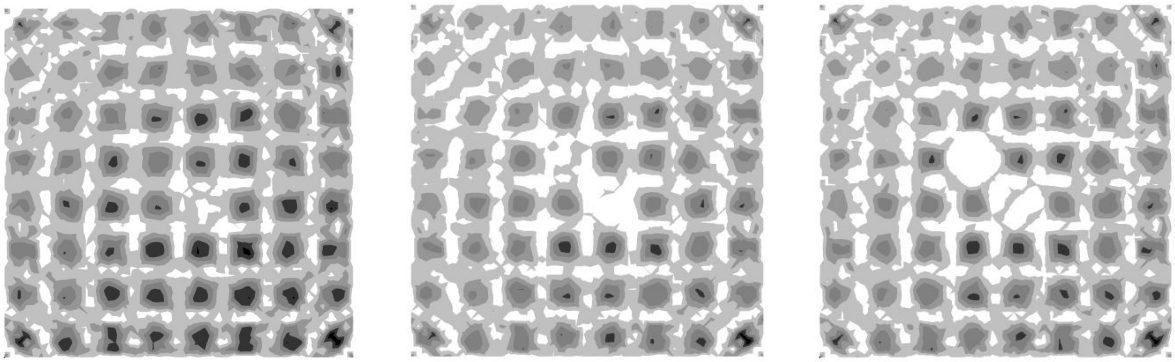

**Figure 9:** Reconstructed images of the complete fuel assembly (left), the assembly with an empty rod in position (E,5) (middle), respectively with a non-active rod in position (E,5) (right).

One may conclude that the removal of a fuel rod may be identified in the reconstructed images. However, in order to further enhance the detection capability, one may consider using dedicated rod-activity reconstructions as well, giving rod activities as illustrated in figure 10.

Empty rod in (E,5)
[relative rod activities]

| | 8 | 7 | 6 | 5 | 4 | 3 | 2 | 1 |
|---|---|---|---|---|---|---|---|---|
| H | 0.775 | 0.775 | 0.767 | 0.896 | 0.902 | 0.909 | 0.903 | 0.921 |
| G | 0.771 | 0.769 | 0.882 | 0.890 | 0.880 | 0.916 | 0.907 | 1.004 |
| F | 0.767 | 0.869 | 0.887 | 0.993 | 0.993 | 1.025 | 1.014 | 1.026 |
| E | 0.887 | 0.875 | 0.984 | 0.605 | 1.025 | 1.047 | 0.980 | 1.018 |
| D | 0.888 | 0.991 | 0.989 | 1.004 | | 0.999 | 1.100 | 1.111 |
| C | 1.000 | 0.994 | 1.003 | 1.133 | 1.135 | 1.141 | 1.114 | 1.112 |
| B | 0.985 | 1.001 | 1.116 | 1.139 | 1.116 | 1.203 | 1.196 | 1.133 |
| A | 1.009 | 1.110 | 1.130 | 1.127 | 1.181 | 1.220 | 1.188 | 1.169 |

Non-active rod in (E,5)
[relative rod activities]

| | 8 | 7 | 6 | 5 | 4 | 3 | 2 | 1 |
|---|---|---|---|---|---|---|---|---|
| H | 0.781 | 0.777 | 0.770 | 0.900 | 0.910 | 0.911 | 0.908 | 0.929 |
| G | 0.773 | 0.765 | 0.884 | 0.884 | 0.879 | 0.910 | 0.905 | 1.011 |
| F | 0.762 | 0.874 | 0.883 | 0.974 | 0.987 | 1.036 | 1.016 | 1.033 |
| E | 0.884 | 0.869 | 0.976 | 0.161 | 1.014 | 1.044 | 0.981 | 1.024 |
| D | 0.887 | 0.989 | 1.001 | 0.982 | | 1.003 | 1.112 | 1.114 |
| C | 0.994 | 0.997 | 1.013 | 1.120 | 1.136 | 1.138 | 1.113 | 1.115 |
| B | 0.986 | 1.000 | 1.126 | 1.135 | 1.121 | 1.215 | 1.198 | 1.132 |
| A | 1.002 | 1.108 | 1.131 | 1.126 | 1.186 | 1.221 | 1.191 | 1.156 |

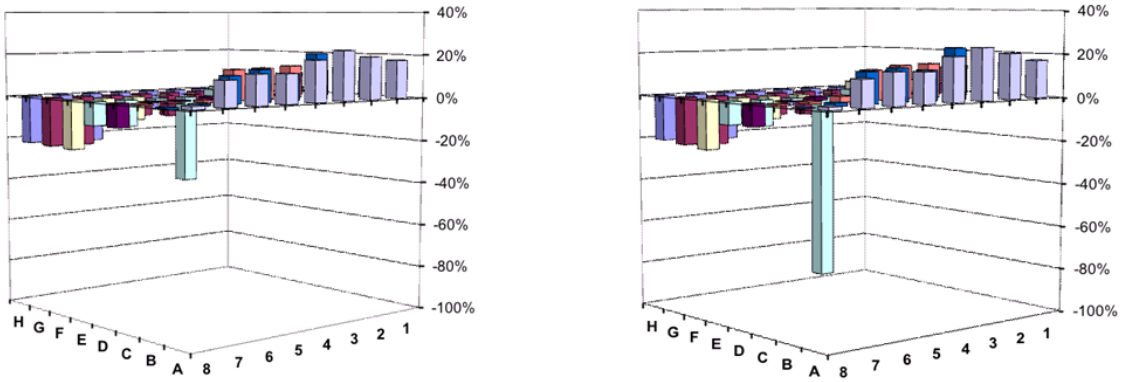


**Figure 10:** Reconstructed rod activities for the assembly with an empty rod in position (E,5) (left), respectively with a non-active rod in position (E,5) (right).

In conclusion, the tomographic technique seems capable of identifying even an individual replaced rod from the centre of the mockup. However, one may expect the level of background in the images to be further reduced if spectroscopic peak analysis with background subtraction is applied, which would further enhance the detection capability. One may also note that the accuracy of the reconstructed activities of the normal rods is better than 1.5% (1 σ), whereas the value in position (E,5) is 84% (non-active rod) respectively 39% (empty rod) below average, which would be detectable in both cases. The difference in response to a non-active rod as compared to an empty rod occurs due to the

difference in attenuation, where the erroneous assumption of having a normal rod in position (E,5) affects the latter case the most.

## 4. Outlook and discussion

One of the primary objectives for future work with the laboratory device is to use it for the verification and possible further refinement of image reconstruction algorithms currently under development. It is also possible to use the device for an optimization of the measurement procedure, taking available time and required activity uncertainties and image resolutions into account.

However, the immediate interest lies in conducting more abundant tomographic measurements and in the following investigations and interpretations of the results. As an example, measurements can be carried out on different rod configurations, allowing for studies of various cases of partial defects where a fraction of the fuel rods have been removed or substituted with non-fuel material.

It is also possible to imagine that other uses of the device may be considered, such as for investigations related to Compton scattering tomography for imaging purposes, where the use of a collimator is redundant. For this specific technique, one relies on Compton scattered gamma-rays emitted by radioactive objects to determine the region of gamma-ray emission. Other possibilities are, for instance, to replace the rod fillings, vary the collimator dimensions or use different detectors (for the latter e.g. $LaBr_3$ has been suggested) in order to obtain properties better suited for other investigations or applications.

Finally, one may note that the time required for a tomographic measurement may govern the usefulness of the technique. For the tomographic measurements presented in section 3.5, using 20 s of measurement time per detector position, the total measurement time was about 12 h using only one detector. With 10 s per position, as in the measurements in section 3.6, the total time was about 6 h, again using only one detector. Using four detectors (preferably with equal collimator slit widths), the total measurement times would have been 3 h and 1.5 h, respectively, which may still be considered relatively long. However, when designing full-scale equipment, shorter measurement time per position (see e.g. [7], where 2 s was used) and more detectors can be envisioned, implying that a total measurement time in the order of minutes may be reached. However, in the laboratory environment discussed here, the somewhat longer measurement times was considered a good tradeoff, allowing for the use of a relatively limited activity in the model rods.

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# 3S (Safety, Security, and Safeguards)-by-Design for Engineering-Scale Pyroprocessing Facility

**Ho-dong Kim, Eun-ha Kwon, Seong-kyu Ahn, Se-hwan Park, Jeong-hoe Ku**

Korea Atomic Energy Research Institute, Daejeon
Nonproliferation System Research Division
Daedeok-daero 989-111, Yuseng-gu, Daejeon, 305-353 Republic of Korea
E-mail: khd@kaeri.re.kr, kwoneh@kaeri.re.kr, skahn76@kaeri.re.kr, ex-spark@kaeri.re.kr, jhku@kaeri.re.kr

***Abstract:***

*Any facility dealing with nuclear materials is subject to a certain level of risk of radiological accidents, sabotage, and proliferation. These risks are managed by domestic regulations and international norms. An expansion of the civil nuclear industry triggered by global warming and the desire for energy security has increased the need for international and domestic resources to meet such obligations. Especially after the September 11 attacks and Fukushima disaster, ensuring the safety and security of nuclear facilities has become essential, along with the safeguard aspect, to the use of nuclear power. Incorporating the 3S (safety, security, and safeguards) concept early in the conceptual design phase of a nuclear facility is important for establishing an effective, efficient, and synergistic system that can reduce safety and/or security risks and proliferation hazards; improve major plant design characteristics, including operational efficiency; and minimize the life cycle cost. In this context, the Korea Atomic Energy Research Institute (KAERI) has developed technologies to enhance safety and security aspects, in addition to safeguards, of advanced (next-generation) nuclear fuel cycles in collaboration with the International Atomic Energy Agency (IAEA) and USA. This paper reviews the concept of 3S-by-design (3SBD): how each element incorporates the other elements of 3S. The paper then describes R&D activities conducted by KAERI on 3S for pyroprocessing, which is spent fuel treatment technology that KAERI has been developing as part of its nuclear energy R&D since 1997, including long-term collaborative work with the IAEA and USA.*

**Keywords:** 3S-by-design; safeguards-by-design; safeguard technologies; pyroprocessing

## 1. Introduction

With the growth and spread of nuclear power, there is a strong need to minimize safety, security, and nonproliferation risks in the design of new nuclear installations throughout the fuel cycle. The 2008 G8 Summit in Chitose, Japan, recognized the importance of nuclear safety, security, and safeguards (3S) toward the peaceful use of nuclear energy. The G8 members reiterated the common interest to continuously improve 3S to ensure a sound basis for international confidence in the sustainable use of nuclear power [1]. Two months prior to the G8 Summit, the International Atomic Energy Agency (IAEA) also noted that “the Agency’s roles in nuclear safety, security, and safeguards complement each other, and all components of 3S’s are essential to the future growth of nuclear applications [2].”

Incorporating the 3S concept early in the conceptual design phase of the nuclear facility is important for establishing an effective, efficient, and synergistic system to reduce safety and/or security risks and proliferation hazards; improve major plant design characteristics, including operational efficiency; and minimize the life cycle cost. By encouraging the design and development of 3S measures early in the project planning and design process, areas with potential conflicts among safety, security, and/or safeguards can also be identified, so they can be resolved at lower cost and minimum schedule impact. Ensuring that the new nuclear infrastructure meets the 3S objectives effectively will require a better understanding of the synergies and conflicts among these three areas.

This paper first reviews the concept of 3S-by-design (3SBD): how each element incorporates the other elements of 3S. The paper then describes R&D activities conducted by the Korea Atomic Energy Research Institute (KAERI) on 3S for pyroprocessing, which is spent fuel treatment technology that KAERI has been developing as part of its nuclear energy R&D since 1997, including long-term collaborative work with the IAEA and USA.

## 2. Concepts of 3S [3]

As identified in the IAEA's "Milestones in the Development of a National Infrastructure for Nuclear Power," nuclear safety deals with the prevention and mitigation of nuclear accidents and the release of radioactivity; nuclear security deals with the prevention, detection, and response to the theft, sabotage, unauthorized access, illegal transfer, or other malicious acts involving nuclear and radiological materials or their associated facilities; and nuclear safeguards deal with the prevention of the spread of nuclear material and weapons technologies.

The domain between safety and security deals with preventing the release of radioactivity. Radioactivity can be released from a nuclear facility owing to system or equipment failure, natural phenomena such as earthquakes, floods, or tsunamis, or unintended human error. This is within the safety domain. Release could also be due to intended human actions such as sabotage, deliberate acts, or external attack. This is within the security domain. The consequences of radioactive release are adverse effects on human health, environmental contamination, and/or devastating damage to the nuclear facility. However, safety and security are not always complementary to each other.

The domain between security and safeguards deals with preventing nuclear weaponization. On the security side, terrorists can steal or forcibly enter a nuclear facility to acquire nuclear material. Similarly, a host country can divert or misuse systems to make nuclear material. The consequence is an undesirable weaponization situation. However, there are ambiguities between security and safeguards such as the motivations of a host country and terrorist group to acquire nuclear material.

The domain between safeguards and safety deals with the inalienable right of non-nuclear-weapons states to the use of nuclear technology as permitted by Article 4 of the Non-Proliferation Treaty (NPT). On the safeguards side, "dual-use controls" designated by the Nuclear Suppliers Group (NSG) and Zangger Committee prohibit applying the technologies to weaponry. On the safety side, the export of nuclear technology is strictly regulated under the NPT articles, which state that the inalienable right to nuclear technology only applies for peaceful purposes. Harmonization between safeguards and safety is essential for nuclear trade and export.

## 3. Pyroprocessing at KAERI

KAERI has been developing pyroprocessing since 1997 to resolve the issue of spent fuel accumulation and produce metal-based fuel for future fast reactors [4]. Figure 1 shows a flow diagram of the process: (1) pretreatment (decladding, voloxidation, and sintering), (2) electroreduction, (3) electrorefining (including uranium recovery), (4) electrowinning, and (5) waste salt treatment. A sodium-cooled fast reactor is being developed in another KAERI project but is not within the scope of pyroprocessing R&D.

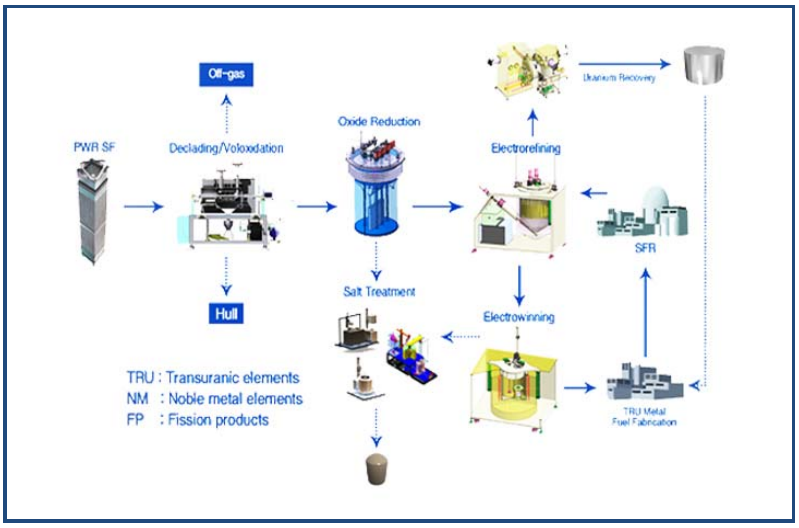


Figure 1. Pyroprocessing flow diagram

In order to demonstrate the integrated pyroprocessing technology, PRIDE (PyRoprocess Integrated inactive DEmonstration facility) has been developed since 2007; it is a cold-test facility to support integrated process demonstration and equipment development, which are essential to realize the pyroprocessing technology. The PRIDE will also support the near-term mission to evaluate and produce reliable data in order to resolve scale-up issues of full-spectrum pyroprocessing technology. The design works for the facility have been completed and the construction and installation works are in near completion now, while the cold operation will be started soon. In PRIDE, the key pyroprocessing technology will be tested and demonstrated using natural and/or depleted uranium with surrogate materials, and the system engineering studies including the design study for facility and equipments, remote operation and maintenance, advanced safeguards and radioactive materials transportation etc. will be performed. The PRIDE consists of a large scale argon-atmospheric cell (40.3 m length, 4.8 m width, 6.4 m height), an argon system, in-cell equipments (transfer lock, in-cell crane, feed through, etc.) and remote operation devices.

*Development of Common Equipment for Safeguards and Security for Pyroprocessing*

To support the nuclear material control and accounting (MC&A) of pyroprocessing, KAERI is actively developing advanced safeguard systems together with Los Alamos National Laboratory (LANL) of the USA. One safeguard approach to pyroprocessing involves the use of gross neutron counting to track the movement of plutonium-bearing material through the facility and processes. Spent nuclear fuel (especially fuel with less enrichment and higher burnup) contains curium, whose isotope $^{244}$Cm is a significant neutron emitter. Accordingly, a gross neutron count of spent fuel counts mostly Cm, and not Pu exclusively. If Cm is transferred with Pu throughout the process, a careful measurement (via destructive analysis (DA)) of the Pu/Cm ratio at the outset can then be used to calculate the Pu contribution to the gross neutron count. Thus, neutron counting can be used to track Pu movement and measure Pu accountability. However, the Cm method is only viable if Cm is completely transferred with Pu in a pyroprocessing facility. Demonstrating that this is the case is a challenge.

A viable MC&A method for plutonium accounting is essential for establishing safeguard requirements for pyroprocessing; it is also important for security in terms of preventing or mitigating insider theft scenarios. In such a scenario, the host country operators need to ensure that plutonium is accounted for during its process and transfer through the pyroprocessing facility.

Similar to the containment and surveillance (C&S) system installed by the IAEA for safeguard purposes, a visualization system of the pyroprocessing facility can be created to detect attempts to steal a significant quantity of nuclear material. The security does not have to expressly depend upon measurement of the nuclear material. In such a case, activity is monitored to observe whether equipment usage, motions, timing, etc., correlate with normal processing (e.g., the material is accounted for) or with abnormal operations (e.g. investigation is needed because material may have been stolen).

## 4. 3S-by-Design for Pyroprocessing

KAERI initiated R&D activities on 3S for pyroprocessing in 2012 when it launched a five-year project called “Proliferation Resistance-Enhanced Pyroprocessing Safeguards and Hot Cell Design Technologies Development.” The primary goal of the project is to develop proliferation resistance-enhanced hot cell design and safeguard technologies for an engineering-scale pyroprocessing facility (10 tons of spent fuel per year) by 2016.

Although both the safety and security aspects of these activities are currently being studied according to their respective legal basis and operation standards, safeguard experts have been involved in the R&D process from the beginning in the early 1990s: nondestructive assay (NDA) measurement equipment, advanced C&S and process monitoring systems, and modelling and simulation of safeguards of pyroprocessing facilities continue to be in progress [5].

Even though the safeguard system for a nuclear facility should be designed and developed at the earliest stage of facility construction, there are no pyroprocessing facilities under international safeguards other than the laboratories for several unit process components. In this respect, safeguard options for pyroprocessing facilities have been evaluated at KAERI since the mid 2000s. In an earlier study, Budlong-Sylvester et al. suggested and evaluated four prospective safeguard approaches for the conceptual design of a pyroprocessing facility: i) neutron balance–Cm accounting, ii) electrorefiner assay, iii) homogenized input, and iv) assay of Pu in spent fuel via Pu/Cm ratio and DA [6]. KAERI then designed a reference engineering-scale pyroprocessing facility and its safeguards system in collaboration with IAEA to support the development of a model facility and its safeguard approaches for future pyroprocessing facilities [8]. The most promising option identified in this study was introducing a homogenized input and using the Pu/Cm ratio. This study also conducted a qualitative safeguardability analysis to investigate the safeguards system and draw recommendations to enhance the performance of the engineering-scale pyroprocessing model facility [8]. For the design features to be able to support safeguard implementation in pyroprocessing facilities, a more effective utilization of the inherent containment and enhancement of portal monitoring and more focus on the accounting material flow into and out of the system will make it possible to achieve the safeguard goal.

Based on these previous studies, a more practical R&D effort is being made by the Safeguards and Security Working Group of the ROK–US Joint Fuel Cycle Study. Technologically, this study has been focusing on the application of existing safeguard techniques and the development of new ones in close cooperation with the process developer and designer. While verification of the Pu and Cm behavior throughout the entire process is a critical target of this study, other aims include the design and evaluation of a comprehensive safeguards system through analyses of the broader issues that impact safeguard aspects of a pyroprocessing facility. A set of safeguard design guidelines and recommendations for future pyroprocessing facilities can be drawn based on these results.

Along with efforts to develop conventional safeguard approaches—which essentially follow current practices relying on material accounting and nondestructive assays—alternative approaches utilizing the concept of safeguards-by-design (SBD) are also being designed and evaluated. Wigeland et al. noted that offsetting the practical limit of material accounting by effective containment, surveillance, and process monitoring measures is more effective [9]. There are several advantages to pyroprocessing operations over conventional aqueous reprocessing operations in this regard. Pyroprocessing operates with a small number of pieces of discrete process equipment and a limited number of material transfers in batch form between unit processes. These transfers are unique in that each batch of process materials only has one destination: either the next process equipment or product storage. All of these operations take place in a concrete hot cell with limited transfer portals. Utilization of the inherent containment and enhancement of portal monitoring will make it possible to achieve the safeguard goal with minimal material accounting measurements by exploiting the nature of pyroprocessing operations. It is reasonable to expect that achieving sufficient transfer monitoring may be more feasible than relying on a combination of various uncertainties from measurements.

## 5. Conclusions

While incorporating the 3S-by-design concept early in the design phase is being emphasized for

commercialization of next-generation nuclear systems, KAERI has dedicated itself to developing technologies that are required to implement an international safeguard system for pyroprocessing, which is a promising and/or advanced nuclear fuel cycle technology in Korea. KAERI is expanding its R&D scope to integrate safeguards with other safety and security objectives in the overall design process of pyroprocessing. The ultimate goal is to achieve an optimum design technology for pyroprocessing that incorporates the enhanced 3S features.

By addressing the concepts of 3S at an initial stage of development, the credibility of the state will be enhanced, which will lead to a pyroprocessing-based fuel cycle being realized in Korea.

## 6. Acknowledgments

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# Experiences with the Development and Implementation of Integrated Safeguards (IS) in some European States

**A. Vincze**

as the Chairman representing the
Implementation of Safeguards Working Group of ESARDA

E-mail: vincze@haea.gov.hu

***Abstract:***

*This paper gives a general overview of the history of and experiences with the development and implementation of new concepts of Safeguards, in particular the IAEA's Integrated Safeguards (IS) in some European States represented in the Implementation of Safeguards Working Group of ESARDA. The major changes in facility safeguards approaches introduced by the implementation of IS are summarized and an analysis of the advantages and disadvantages experienced by the States is made.*

*Most of the information contained in this paper has been discussed in the course of the "round table updates" delivered by each member over many years in the ESARDA IS WG meetings. The European Commission's Safeguards Services, who implement safeguards jointly with the IAEA in the EU, are also permanent contributors to the WG's activities. EURATOM also acts as channel of communication between the IAEA and EU national authorities and operators, thus facilitating the implementation of IAEA IS in the EU territory.*

*The group decided to collect this information on history of and experiences during the implementation phases of the IAEA's Integrated Safeguards (IS) in a structured way by means of specially developed State information sheets. They serve as a tool for information compilation thereby creating a solid base for safeguards implementation studies to be carried out by the IS WG. This paper is one example of such studies.*

**Keywords:** Safeguards Implementation; Integrated Safeguards; State's experience; EURATOM's role;

## 1. Introduction

The implementation of IAEA's Integrated Safeguards (IS) in the Non-Nuclear Weapon Member States of the European Union was completed in 2010. To mark this significant achievement in Europe, the Integrated Working Group of ESRADA considered that it had completed its mandate and decided to change its name to Implementation of Safeguards Working Group, redefining also its terms of reference to a broader implementation scope [1]. As part of this transformation, the group also decided to summarise its findings in connection to the implementation of IS in Europe and for that purpose a "State information sheet" was developed to be filled in by WG members voluntarily.

The information sheet gives a general overview of the history of and experiences with the development and implementation of the IS in a given State. Special focus is given to the development of the State's nuclear fuel cycle and the SSAC in connection with the key safeguards milestones. The major changes in facility safeguards approaches introduced by the implementation of IS are summarized and a short analysis of the advantages and disadvantages of IS implementation experienced by the State is given.

Most of the information contained in these information sheets has been outlined in the course of the "round table updates" delivered by each member over many years in the ESARDA IS WG meetings. The aim of the compilation of these sheets is to collect all these information updates in a structured way to form a solid base for future safeguards implementation studies to be carried out by the IS WG.

In the following sections, the conceptual framework of IS will be briefly overviewed focusing on the implementation aspects of IS and to assess the efficiency and effectiveness.

So far information sheets from Belgium, Czech Republic, Estonia, Finland, France, Germany, Hungary, Norway, Spain, Sweden and Switzerland have been submitted to the group. France is a Nuclear Weapons State and therefore IS implementation experience is not given. in Switzerland the broader conclusion has not yet been drawn by the IAEA, but the Safeguards implementation has changed during the last years.

In the subsequent sections the main safeguards characteristics of the States contributing to this paper will be given, together with their experiences in connection to the implementation process. Also for some cases the change in the facility level safeguards approach and the resulting change in IAEA effort (e.g. inspection days) together with change in the effort of SSAC/Euratom/Facilities will be discussed. At the end of the paper the role of SSAC/Euratom in the implementation process will be outlined and the main points of the paper summarized.

## 2. Conceptual Framework of IS

For States with Comprehensive Safeguards Agreements (CSA) and Additional Protocols (AP) in force, where no indication has been found by the IAEA that any undeclared nuclear material and activities are present, the Secretariat can draw the broader conclusion that all nuclear material in a State has remained in peaceful activities. Subsequently, the IAEA implements an integrated safeguards approach for that State.

With the introduction of integrated safeguards for a given State, the combination of the traditional safeguards regime (CSA) and AP have to be integrated in such a way that both the economic costs of implementation (efficiency) and the effective supervision of a country's nuclear materials and activities are optimal. This could only be done having a State level conclusion of the absence of undeclared nuclear material and activities in a State which can justify the redistribution of resources needed to meet the facility level Safeguards Criteria. The criteria are based on the assumption that clandestine activities needed to further process diverted materials are present in the State. Since the broader conclusion challenges this assumption, emphasis could be directed to the coverage of the viable acquisition paths in that State by measures implemented under the AP, whereby the intensity of inspection activities at declared facilities and LOFs can be reduced.

After drawing the broader safeguards conclusion for a State as whole, the State-level integrated safeguards approach is designed, discussed with the State and applied. Such an approach should be developed on a non-discriminatory basis, using safeguards verification objectives common to all States. The approach should also take State specific factors into account, such as the effectiveness of the State (or regional) system of accounting for and control of nuclear material (SSAC/RSAC) and the features of the State's nuclear fuel cycle. The development of IS also aims at avoiding undue burden on SSAC/RSAC and the facilities with the optimal combination of safeguards measures to efficiently and effectively drawing conclusion on non-diversion of declared nuclear material and the absence of undeclared nuclear material and activities.

The implementation of IS is evaluated in this paper by the assessment of:

- the efforts devoted to the design, negotiation and establishment of IS, which is roughly proportional to the time elapsed between the broader conclusion and the implementation;
- the resulting change in IAEA effort (e.g. inspection days given in person days of the inspection (PDI) or the calendar-days in the field for verification (CDFV));
- the change in effort of SSAC/Euratom/facility operators;

In the following sections the main safeguards characteristics and IS implementation experiences of the States contributing to this paper is summarized based on their information sheets. Also the parameters described above will be assessed for some of the States as case studies and in general for all non-nuclear weapon states.

## 3. States` safeguards characteristics and experience

The States contributing to this paper represent a broad distribution of fuel cycle types, as shown in Table 1. The main safeguards milestones are summarized in Table 2, where the time needed for the broader conclusion since the AP came into force, as well as the time used for the implementation of integrated safeguards after the broader conclusion, can be seen.

| | BEL | SWI | CZE | ESP | FIN | FRA | GER | HUN | SWE | NOR | EST |
|---|---|---|---|---|---|---|---|---|---|---|---|
| NPT status | NNWS | NNWS | NNWS | NNWS | NNWS | NWS | NNWS | NNWS | NNWS | NNWS | NNWS |
| Minig/milling | N | N | 1+1 clsd | clsd | 1 | N | N | clsd | N | N | N |
| Conversion | N | N | N | 1 | N | 1 | N | N | 1 | N | N |
| Enrichment | N | N | N | N | N | 2 | 1 | N | N | N | N |
| Fuel Fabrication | 1 | N | N | 1 | N | 3 | 1 | N | 1 | 1 | N |
| Power Reactor in operation | 7 | 5 | 6 | 8 | 4 | 58 | 9(8)* | 4 | 10 | N | N** |
| Research Reactor | 3 | 3 | 3 | 1 | 1 | | 8 | 2 | 2 | 2 | N |
| Reprocessing | N | N | N | N | N | 1 | N | N | N | N | N |
| Offsite SF Storage facilities | N | | N | 4 | N | | 3 | 1 | | N | N |
| LOFs/CAMs | 2 | 1 | 170 | 4 | | | 40 | 1 | 20 | 50 | 1 |

**Table 1.** Nuclear Fuel Cycle of contributing states. * Following the Fukushima accident in spring 2011, the 8 oldest of the then 17 operating reactors in Germany were shut down in March 2011 and will not resume operation. The remaining reactors will be all shut down by 2022.; ** (2 reactors on submarines are decommissioned)

| State | BEL | SWI | CZE | ESP | FIN | FRA | GER | HUN | SWE | NOR | EST |
|---|---|---|---|---|---|---|---|---|---|---|---|
| NPT-ratification | 1968 | 1977 | 1968 | 1987 | 1969 | 1992 | 1975 | 1968 | 1970 | 1972 | 1992 |
| Comprehensive Safeguards Agreements (CSA) in force | 1975 | 1978 | 1972 | | 1972 | NA (INFC. 290 1981) | 1977 | 1972 | 1974 | 1972 | 1997 |
| AP in force | 2005 | 2005 | 2002 | 2003 | 2004 | 2003 | 2004 | 2000 | 2004 | 2000 | 2005 |
| Broader conclusion | 2007 | Not drawn | 2006 | 2007 | 2007 | NA | 2008 | 2003 | 2007 | 2002 | 2007 |
| IS implemented | 2010 | NA | 2007 | 2010 | 2008 | NA | 2010 | 2004 | 2009 | 2002 | 2009 |

**Table 2.** Safeguards milestones

It can be seen that in most of the cases presented in Table 2, the time needed for the IAEA to draw the broader conclusion after AP has come into force in a State varies between 2-4 years and does not directly depend on the extent of the fuel cycle of the State. For example, in the case of both the Czech Republic and Germany 4 years were needed for the IAEA to draw the broader conclusion, despite the great difference in the two fuel cycles (see Table 1). For Switzerland eight years has already elapsed since the AP came into force (as per 1. May 2013). The reason of this delay will be explained under the country specific history. Several issues that emerged during the AP implementation have been already addressed in a previous paper [2] and will not be mentioned here.

After the broader conclusion, the implementation of IS takes on average 1-2 years, in line with the original expectations by the IAEA [3].

Next, a brief overview of the States' experiences is given with special attention of the main obstacles and/or good practices encountered during the implementation phase of the IS.

### 3.1. Belgium

The NPT was signed by the Kingdom of Belgium in 1968, and the multilateral safeguards agreements were signed in coordination with the other NNWS members of the European Atomic Energy Community (Euratom) and Euratom in 1973, and approved by the law of 14$^{th}$ March 1975. The law of 20$^{th}$ July 1978 provides the conditions allowing the IAEA inspectors to carry out activities of monitoring and verification under the safeguards agreements. The additional Protocol was signed on the 22$^{nd}$

September 1998 together with the twelve other NNWS members of Euratom and the two NWS. The application law was enacted the 1st June 2005. According to the annex III of the Additional Protocol, Belgium decided to entrust to the Commission of the European Communities implementation of certain provisions which under the AP are the responsibility of the State and became thus a " side letter state".

In Belgium, the Federal Agency for Nuclear Control supports international inspectorates (Euratom and IAEA) for the accomplishment of their monitoring and verification missions under the Safeguards Agreement and its Additional Protocol. According to the law of 20th July 1978, IAEA Safeguards inspections have to be performed at the same time than and in conjunction with Euratom inspections (art. 2). During inspections, the inspectors of the IAEA and Euratom can be accompanied by agents of the Federal Agency for Nuclear Control (art. 10).

Like any new process, IS implementation in Belgium encountered some practical difficulties at the beginning, which were solved after a while: the limited preparation time for the operator in case of RII and SNRI inspections led in some cases to organisational difficulties at the side of the operator (availability of equipment and human resources). But thanks to organisational changes and flexibility from the operator part, it was possible to accommodate with these practical difficulties.

### 3.2. Estonia

The NPT came into force in 1992 and the application of safeguards in Estonia under the NPT is in force since 24 November 1997 as INFCIRC/547. It was suspended on 1 December 2005, on which date Estonia had acceded to the agreement of 5 April 1973 between the non-nuclear weapon States of Euratom, Euratom and the IAEA (INFCIRC/193). Similarly, the AP came into force as INFCIRC/193 Add.8 IAEA-ESTONIA (Add. Prot.) on 1 Dec 2005. The broader conclusion was drawn in 2007 and IS were implemented on the 1st of May 2009. Estonia does not have nuclear power plants, research reactors nor intermediate or high-level waste containing plutonium, enriched uranium or uranium-233. Estonia has two shutdown nuclear submarine reactors compartments in the Paldiski site, former USSR nuclear submarine training centre, which is in the process of decommissioning. The implementation of IS gave no rise to problems due to the limited fuel cycle in the country.

### 3.3. Germany

The Federal Republic of Germany signed the Non Proliferation Treaty (NPT) on November 28, 1969 and acceded to the NPT on May 2nd, 1975. The INFCIRC/153-type safeguards agreement with the IAEA is the trilateral agreement signed by the non-nuclear weapon Member States, the Community and IAEA on the 5th of April 1973 (INFCIRC/193) which entered into force in the community on the 21st of February 1977. Germany has, together with all then 15 EU States, the Additional Protocol (AP) entered into force in the EU on the 30th of April 2004 as INFCIRC/193/Add. 8.

Germany, being a side-letter state, has no specific national safeguards authority; the competent national point of contact for safeguards issues is the Federal Ministry of Economics and Technology.

In the autumn of 2004, 180 days after the entry into force of the AP, the initial declaration for Germany was submitted to the IAEA via the Commission. In the following years, there was a phase of intensive information exchange with the IAEA where especially activities carried out in the past were a subject of investigation. Several times, Germany received an official request from the IAEA for amplifications or clarifications and many times exchange of information took place in a less official way on the working level. In the period until the implementation of IS the IAEA conducted around 40 Complementary Accesses in Germany. The Commission was always involved in all these steps and also served as the communication channel to receive and transmit information. For 2008 the IAEA was able to draw the broader conclusion and in May 2009 the IAEA presented to the German officials and the European Commission their concept for IS implementation in Germany. IS was finally implemented in Germany as of the 1st of January 2010.

### 3.4. Sweden

Sweden signed the Non-Proliferation Treaty in 1968 and ratified it in 1970. The INFCIRC/153-type safeguards agreement between Sweden and IAEA INFCIRC/234 entered into force in April 1974. Sweden became a member of the European Union on January 1st 1995 and hence the agreement was

replaced in 1995 by the tripartite agreement between the non-nuclear weapons European Member States, the Community and IAEA, INFCIRC/193. Sweden has, together with all then 15 EU States, concluded an Additional Protocol (AP) and remained a 'non-side-letter' state. The broader conclusion was drawn by the IAEA in 2007 and IS were introduced in Sweden from January 15th, 2009.

Sweden decided to have the national authority, SSM, as the common site representative for all Swedish sites and thus all communication with the IAEA and EU-Commission concerning AP is done through the SSM. SSM participates in all IAEA inspections/accesses in Sweden according to a governmental decision. One important step was to create an organization for preparedness for complementary access, short notice and unannounced inspections, both at the facilities, as well as at the national authority.

In the beginning of IS implementation problems were mainly related to lack of knowledge and experience of IS from all parties involved. Also communication problems occurred resulting in errors in inspection notifications.

### 3.5. Norway

The Non-Proliferation Treaty entered into force in Norway on the 5th of March 1972 and the bilateral Safeguards Agreement of INFCIRC/153-type (full-scope) INFCIRC/177/ IAEA-Norway was concluded in the same year (1 March 1972). The Additional Protocol (AP) came into force on 16 May 2000 (INFCIRC/177/add.1/ IAEA-Norway). The implementation of the AP caused some major changes in the inspection regime. Unannounced inspections and complementary accesses were introduced. Integrated Safeguards were implemented in Norway in 2002. Norway was at the time one of few countries in the world to have implemented this regime fully [3].

As a result of the signing of the AP a new regulation for small users of nuclear materials was issued 12 May 2000. The implementation of the AP added momentum to the process to move the SSAC from the operator to the State authority (NRPA), and thereby to make the SSAC more independent of the operation of the nuclear facilities and the numerous small users. The process was completed 12 June 2004

The transition from traditional safeguards to integrated safeguards has been rather smooth, though there have been some issues. In 2008, the IAEA insisted on verifying spent fuel during UI, as if they wanted to return to the old fashioned regime. This was outside the UI procedures that had been agreed. The NRPA responded to this increased inspection effort in a letter to IAEA. The IAEA admitted that this was not as agreed, and resumed to the agreed UI procedures.

### 3.6. Finland

Finland signed the Non-Proliferation Treaty in 1968 and ratified it in 1969. The bilateral Safeguards agreement of INFCIRC/153-type (full-scope) was put in force in 1972 with country code of INFCIRC/155 IAEA-Finland. As all then 15 EU States, Finland became member of the EU in 1995 and signed the INFCIRC/193/Add. 8 agreement, so far remaining a non-side letter state, on the 22nd of September 1998. The AP entered into force in the EU on the 30th of April 2004. Broader conclusion was drawn for Finland in 2007 and IS was implemented as of 15.10.2008.

Due to the changes in safeguards implementation, amendments in nuclear energy legislation was necessary. Finland has decided that STUK would be the site representative for all sites. According to the regulation, STUK must participate in the all IAEA and Euratom inspections in Finland. STUK has organised several information meetings on the effects of integrated safeguards for the nuclear power plant operators and preparatory trilateral meetings with the IAEA and the Commission since 2007. The implementation procedures were reviewed at four meetings held in Vienna 12 February, 24 June, 5 November, and 10 December 2009, and one meeting held in Luxembourg 17 March 2009. In the meanwhile, the operators updated their handbooks in order to incorporate the necessary actions for the new types of inspections. Since then, one or usually two trilateral meeting have been arranged annually.

To facilitate smooth performance of inspections, STUK has created "SNUICA" system (Short Notice and Unannounced Inspections, Complementary Accesses). SNUICA alerts come via STUK's

emergency preparedness system (24/7) and there is always a safeguards inspector on duty and ready to go. This ability guarantees STUKs presence in all inspections. The scheme has been improved lately by starting co-operation with STUK's on site NPP safety inspectors who are available at the beginning of inspection until the safeguards inspector on duty arrives.

In Finland, IS implementation has been very successful. There has been some discussion about SNRI inspection notice timelines, but it really hasn't made any difference if notice time is 24 h or 48 h. Our experiences have shown that active co-operation between all parties (Finland, IAEA, EC) is fruitful and effective way to implement safeguards in Finland.

### 3.7. Hungary

Hungary, signing the NPT among the first States in 1969, signed a comprehensive Safeguards Agreement with the IAEA on the verification regime of the NPT obligations, which entered into force in 1972. The AP was signed by Hungary in 1998 and entered into force in April 2000. At that time, out of the States with considerable nuclear industry, only Japan, Australia and Canada had Additional Protocol in force. The initial Additional Protocol declaration of Hungary was sent to the IAEA in 2000. Hungary, together with Finland and later United States, also offered to test procedures for complementary access activities and offered support to the IAEA in arranging training for IAEA inspectors on complementary access at nuclear sites and locations.

The broader conclusion was drawn in 2003 and IS were introduced already in 2004. Difficulty arose however, with the implementation of IS due to the incident that took place in 2003 in Unit 2 of NPP Paks, whereby 30 irradiated fuel assemblies were destroyed during cleaning operation. IS therefore could not be introduced on Unit 2, where special safeguards measures were put in place for a long period of time. Apart from this exception, the new IS facility approaches for the other MBAs in the country were introduced without significant difficulties. The change of scheduled interim inspections with a smaller number unannounced inspection and/or short notice random inspections required the SSAC to have a safeguards inspection on duty who can immediately be mobilised if necessary.

After Hungary became member of the European Union in 2004, the multilateral safeguards agreement between the IAEA, the Euratom and the Member States, including its AP, came into force in Hungary on the 1st of July in 2007, suspending the bilateral agreements.

### 3.8. Czech Republic

Non-Proliferation Treaty was signed and ratified in 1968 and the bilateral Safeguards Agreement was concluded by Czechoslovakia in 1972 as INFCIRC/173 IAEA-CSSR. After splitting Czechoslovakia in 1993, it was replaced by INFCIRC/541 IAEA-CR in 1997. Additional Protocol was signed by the Czech Republic in 1999 and ratified in 2002 as INFCIRC/541 add.1 IAEA-CR (Add. Prot.). Broader conclusion was drawn by the IAEA in 2006 and implemented IS in 2007. After the Czech Republic joined the EU in 2004, it acceded to the multilateral INFCIRC/193 and its AP, remaining a non-side letter state in 2009.

First declaration according to article 2.a.(ix) of AP was sent to the IAEA in November 2002. The initial declaration, which contained 35 declarations, was submitted to the IAEA in December 2002. The broader safeguards conclusion was approved by DG-IAEA in April 2007. The State Level IS Approach was approved by DDG-SG in May 2007 and Implementation of a State Level IS Approach started from 18 June 2007.

After signing the Additional Protocol, the Czech Republic was giving full assistance to the collaboration with the IAEA. Even before its entry into force, the competent national authority, SONS, in cooperation with the Agency, organized a series of seminars, the aim of which was to get the small holders of nuclear materials acquainted with the new provisions of the Additional Protocol. The initial declaration contained relatively big number of the sites, in principle all LOFs where nuclear material was consumed were described as sites.

After submitting the first declarations to the IAEA, the Agency's inspection efforts have raised. The findings of the inspections in some cases triggered further actions by the Agency that included requests for additional information or access to the locations in question. The CA was performed

mainly with 24 hr advance notification at sites, locations connected with possible exports of trigger list items or operations specified in Annex I to the AP; in one case the CA was aimed to resolve some questions from the history.

Although the IAEA used surveillance systems at both NPPs even before 2007, the servers are now connected to the internet and allow remote transfer to Vienna and Luxembourg. This allows the IAEA to reduce verification activities during refuelling. Different number of the IAEA inspections during loading of the containers CASTOR at both NPPs is done by the possibility of the NPP Dukovany operator to attach and detach the VACOSS seal, connected to RM, during the spent fuel transfers. At the NPP Temelin the IAEA inspectors verify the spent fuel by the Ion Fork detector during CASTOR loading and are therefore present for the installation of FDET, CASTOR verification and CASTOR sealing. In the future sealing procedure during CASTOR transfers will be adopted in both NPPs, as well as the Ion Fork measurement.

## 3.9. Spain

Spain ratified the NPT in 1987 and when it became member of the EU in 1986 it signed the INFCIRC/193/Add. 8 agreement as a side letter state. The Additional Protocol entered therefore into force in Spain on April 30$^{th}$ 2004. Within 200 days the first declaration concerning the period from 01/05/2004 to 31/06/2004 had to be submitted to the European Commission, including export declarations. For the implementation of the AP, Spain is a side-letter state.

It was in 2004 when the IAEA first announced to the State authorities and the operator of the Spanish Fuel Fabrication Plant (FFP) its wish to introduce the Short Notice Random Inspection (SNRI) concept in this type of plant. Negotiations on the implementation procedure endured up to the end of 2006, when all the parties agreed to put to test the scheme in a six-month field trial (from the 1st of March to the 1st of September 2007).

Although there never was any official assessment of the trial neither from the IAEA nor from Euratom, by the end of July 2007, a meeting between the IAEA, Euratom, the State and the operator was held to have a preliminary discussion on the results of the trial and to agree on the way ahead for the rest of the year. All parties concurred that the scheme foreseen in the field trial had gone well, without noticeable difficulties and with satisfactory results. Only minor issues regarding the exchange of information and some improvement in the interface between IAEA and Euratom for triggering the inspections were identified as matters for further development. Given the fact that for the rest of the year no routine inspection had been planned, it was agreed to extend the trial, under the conditions agreed, until the following PIV or until a revised procedure was available for permanent implementation of the SNRI scheme at the Spanish FFP.

Negotiations were resumed in early 2008 and lasted until January 2009. At the same time discussions between Euratom and the Agency on the practical arrangements for IS implementation in EU Member States took place, aiming to produce generic Facility-Type Partnership Approach papers as a basis to issue more detailed facility specific papers on which the particularities of each installation were taken into account. In the case of Spain, the decision was taken to issue a specific paper covering all the NPPs and another for the FFP.

During the last round of negotiations, the main topics of discussion for the FFP were the information to be reported in the mail-box daily notification system and the retention time (time between the receipt and processing of items or their production and packing for shipment), the latter being the hottest issue to be addressed.

The Specific Partnership paper was finally signed by the IAEA and the European Commission (EC) and officially distributed among all the involved parties by mid July 2009. On September 1st, 2009 the new system entered into force at the Spanish FFP.

In a similar manner, after almost two years of intense meetings (between IAEA, Euratom, National Authority and operators), Spain has a Facility Specific Safeguards Approach for LWRs since 2009. This document describes the general implementation arrangements under integrated safeguards for the Spanish LWRs and, in its Annex V, the specific arrangements by facility. These arrangements have been followed by inspectors since the IS has been implemented by the 1$^{st}$ of January of 2010.

Inspections have been developed satisfactorily, apart from problems of minor importance as some inspectors have not taken into account the aforementioned documents and, consequently, it was difficult to carry out the specific inspection arrangements agreed under IS.

On the other hand, some permanent cameras installed in the Spanish LWRs need to be maintained quite regularly, what is not along the lines of reducing the number of inspections under IS. Remote data transmission may resolve this issue.

### 3.10. Switzerland

As in many other countries after entry into force of the AP, the IAEA confronted Switzerland with many issues and clarifications digging also in the pre-NPT history of the country. Among these of special mentions are: AP site definitions of several nuclear facilities, pre-AP environmental samples with peculiar results, past activities devoted to the study of a possible nuclear weaponisation of Switzerland and the relations between Switzerland and Liechtenstein concerning export control. All issues but the last could be resolved, leaving Switzerland up to now without broader conclusion

Since 1923 Switzerland has a custom union treaty with Liechtenstein. For the implementation of this treaty Liechtenstein has for import and export matters the same status as the others Swiss cantons. The Swiss authorities are responsible for imports and exports to and from Liechtenstein. The Swiss custom border encompasses also the territory of Liechtenstein. Movements of goods between Switzerland and Liechtenstein are legally considered as domestic transports and there is no custom border between Switzerland and Liechtenstein.

Since Liechtenstein and Switzerland have both independently signed the comprehensive Safeguards agreement, the IAEA consider both states as completely separated entities. Switzerland ratified the AP, but Liechtenstein found out that in doing so movements between Liechtenstein and Switzerland of goods listed in Annex II would have to be declared under Article 2a.(ix)(a). Liechtenstein pointed out that it will not ratify the AP before this issue is solved. There are currently no legal means in place to have these movements declared. Insofar meetings between both states and the IAEA were inconclusive since the IAEA is not even considering the possibility of some possible “microscopic” adjustments to the agreements.

Nevertheless the IAEA approach has evolved during the last years taking into consideration Switzerland’s efforts and improvement in the Safeguards national system. Among others Switzerland was one of the first country to have introduced Remote Monitoring in LWRs and in one storage facility (2000-2001) This factor and others combined led for example to a more relaxed timing of inspections at LWRs and reduced inspections at two storage facilities. Meanwhile a new inspection regime for LWR is being elaborated to reduce here too the number of inspections. In conclusion it could be argued that in one or two years Switzerland will have in practice an IS approach without getting the formal recognition of the broader conclusion.

## 4. Facility safeguards approaches and efficiency change

While under the IS the safeguards objectives remain similar for all States, the individual State Level Approaches may differ, since they are intended to reflect the optimal combination of safeguards measures for each State, taking into account State specific factors and adjusting safeguards intensity accordingly, for example, through specifying selected safeguards measures and similar type of facilities to be grouped for inspection. A State-level integrated safeguards approach is formulated for a State by combining the integrated safeguards approaches for the specific facility types present in the State and the implementation of measures of the Additional Protocol (CAs) taking into account the nuclear fuel cycle and interaction between facilities in order to cover the plausible acquisition paths in a State.

Model facility level integrated safeguards approaches have been developed by the IAEA for the typical facility types with some optional flexibility to account for differences in State-specific characteristics. As an example for Light Water Reactors (including MOX fuel), the applied safeguards approaches before and after are shown in the case of some States in Table 3.

| LWR with MOX fuel - Germany | |
|---|---|
| traditional SG | IS-approach |
| • Annual PIV (Pre-PIV, Core Control, Post-PIV)<br>• Monthly Inspections for fresh MOX fuel<br>• Quarterly interim inspections<br>• Surveillance to cover time between inspections (continuity of knowledge)<br>• Core sealing | • Annual PIV (Pre-PIV, No Core Control, Post-PIV)<br>• Quarterly Inspections for fresh MOX fuel<br>• Short Notice Random Interim Inspections SNRI) with 20% selection probability per year/reactor (24 h notification)<br>• Surveillance in overwriting mode to cover open core and RII announcement period<br>• Core sealing |
| **LWR without MOX fuel - Hungary** | |
| traditional SG | IS-approach |
| • Annual PIV/DIV/year/MBA (Pre-PIV, Core Control, Post-PIV)<br>• Sealing of reactor core<br>• Continuous surveillance in overwrite mode<br>• Verification of SF shipments<br>• Three interim inspections/year/MBA | • Annual PIV/DIV/year/MBA (Pre-PIV, No Core Control, Post-PIV)<br>• Sealing of reactor core.<br>• No sealing on SF pool.<br>• Surveillance in overwrite mode + additional temporary backup surveillance<br>• No verification of SF shipments (to be verified at the receiver MBA)<br>• One SNRI/year/Facility (24 h notification) |

**Table 3.** Comparison between Safeguards approaches before and after IS implementation in LWR.

As can be seen from Table 3, by introducing the IS facility approach for LWR reactors, especially with the MOX fuel, the number of in-field inspections can be reduced with the same effectiveness, as the traditional safeguard approach. This is indeed justified for the IAEA by having the broader conclusion for a State and also by the technical development of C/S safeguards technology.

This is illustrated below for some of the States showing the change of in-field inspection efforts in PDI units.

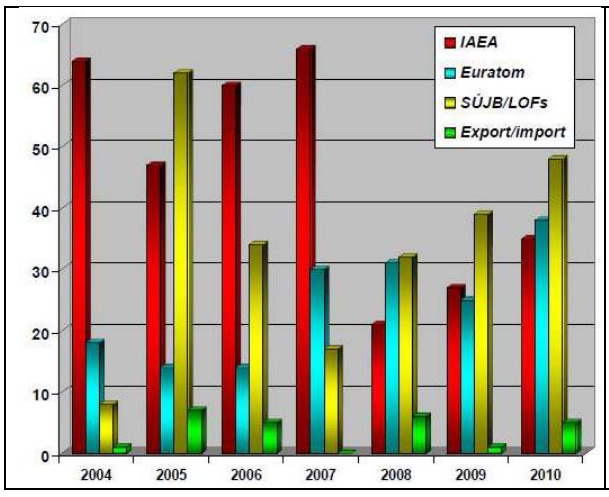


**Figure 1/A.** Czech Republic: IS introduced January 2007

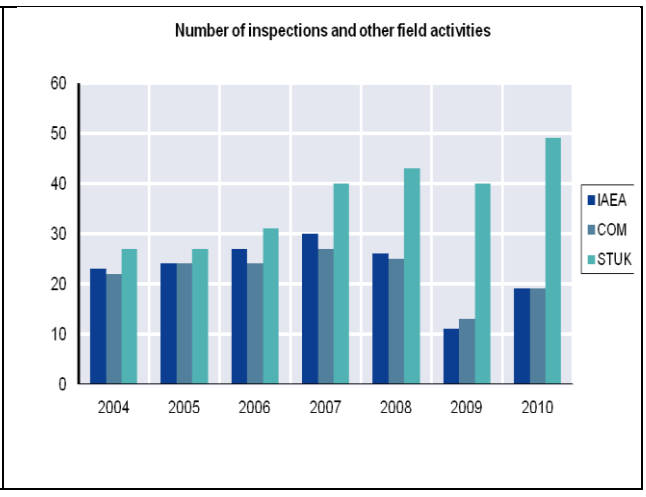


**Figure 1/B.** Finland: IS introduced in 2008

As shown by Figure 1/A, for the Czech Republic the inspection effort of the IAEA decreased after IS implementation in 2007. The frequency for PIV inspections reduced for MBAs WCZD, WCZF, WCZG and WCZV (smaller RRCA or other types of small installations). The announced interim inspections

were practically eliminated and were replaced by a limited number of unannounced and random interim inspections with 24 hour advance notification. The Euratom inspectors continue with the verification activities as before the adoption of the IS approach, while the state (SONS - SÚJB) inspectors focus more on the inspections at the small holders of nuclear material (LOF MBA). They also accompany the IAEA and Euratom inspectors by law and perform some inspections concentrated on the verification of stated usage of imported trigger list and dual-use list items and in some cases on verification of these items before the SONS issues the export licence.

According to Figure 1/B, there was a significant change in 2009 of the inspection efforts for the IAEA and Euratom as expected according to the new IS for Finland. The increase in 2010 was owing to first inspections to the Geological Repository, additional inspections to the Loviisa NPP due to operational reasons, and higher number of random SNRIs to the spent fuel storages. The number of STUK inspection days increased, because of increased number of safeguards and security inspections to small holders and the Research Reactor.

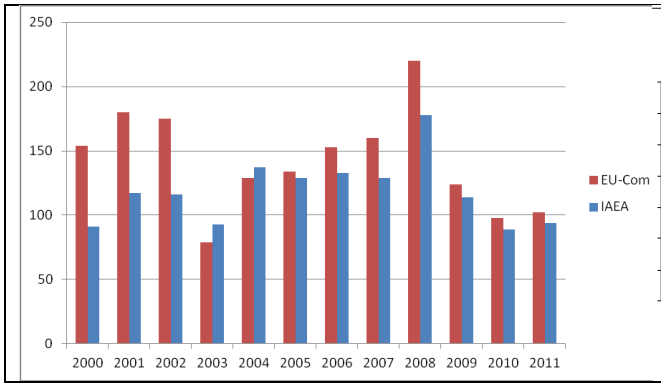


**Figure 2/A.** Sweden: IS introduced January 2009

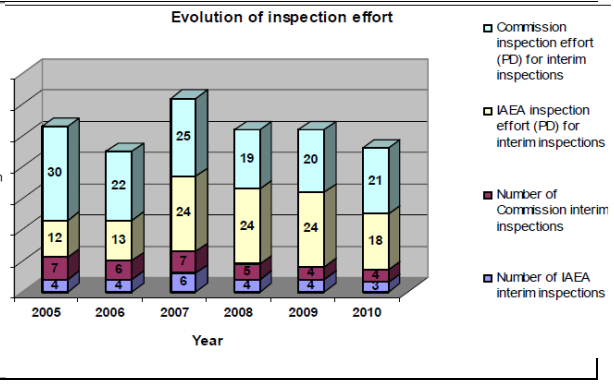


**Figure 2/B.** Spain: Fuel Fabrication Plant inspection efforts. 2007-2009 Trial period, IS since 2010

In Sweden the high figure in 2008 was caused by failed PIV at the fuel fabrication plant. Since the AP's entry into force in 2004 there have been 37 CAs and SNRIs (incl UIs). For power reactor there are no real reductions so far in inspection effort mainly due to the introduction of pre- and post-PIVs and unfortunately the status of the NPPs. For the fuel fabrication plant there are fewer inspections. Studsvik research facility and the spent fuel storage have experienced a rough 50% reduction.

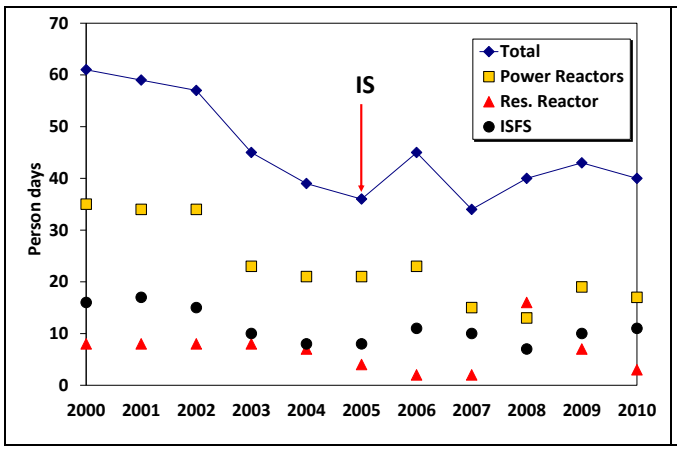


**Figure 3/A.** Hungary: IS implemented in 2004

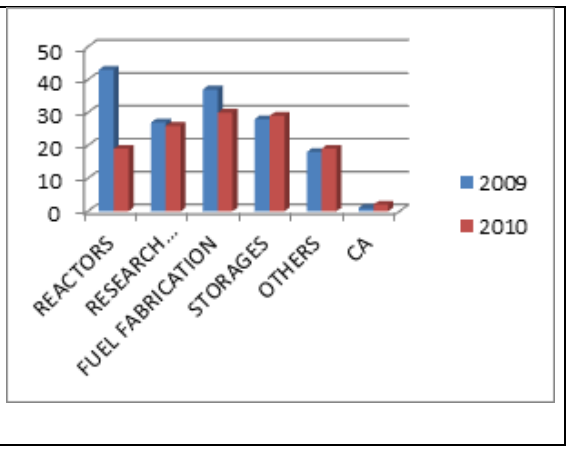


**Figure 3/B.** Belgium: IS implementation in 2010

From the analysis of the trend of the annual inspections carried out in Hungary for the Paks Nuclear Power Plant, the Budapest Research Reactor and in the Interim Spent Fuel Storage Facility (ISFS), it

can be seen that the person-days of inspections have somewhat decreased since 2003 as compared to the levels in 2000. It can also be concluded that the introduction of the integrated safeguards system in Hungary did not result in a decrease of the amount of the IAEA inspections. This is, however, not due to inefficiency of the new system, but to the extra inspection activities required by the 2003 serious incident when fuel assemblies were damaged at the Paks Nuclear Power Plant, and by the repatriation of highly enriched fuel from the Budapest Research Reactor to Russia in 2008.

In Belgium significant decrease of inspections occurred only in the case of reactors and fuel fabrication plants.

The trend of in-field effort of IAEA and Euratom for the 25 NNWS of the EU is shown on the following figure:

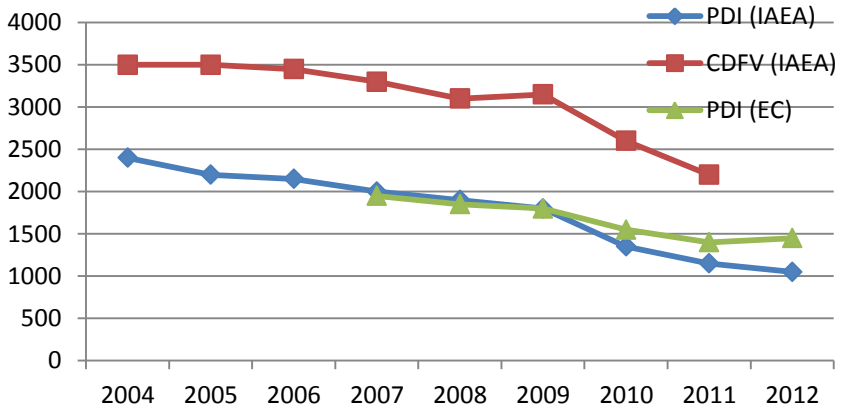


**Figure 4.** Data presented to the group by W. Kahnmeyer, EC, DG ENER-E4, at the meeting of the ESARDA WG on Implementation of Safeguards, Bergen, 10-11 April 2013

It can be seen from Figure 4, that the total IAEA in-field effort (combined inspections and CA, shown as person days of inspection – PDI and total days in-field - CDFV) has been clearly reduced for the 25 NNWS of the EU, which is justified by the broader conclusion for each State and also by the safeguards technology development. The Euratom in-field safeguards effort (inspections only, shown as person days) has also been reduced but to a lesser extent. This reduction is mainly due to technological development in safeguards means (extensive use of RDT) and the introduction of common IS approaches based on the facility specific new partnership approaches with the IAEA.

The general trend is also true globally as shown in Figure 5., which was compiled on the basis of the IAEA annual safeguards statements.

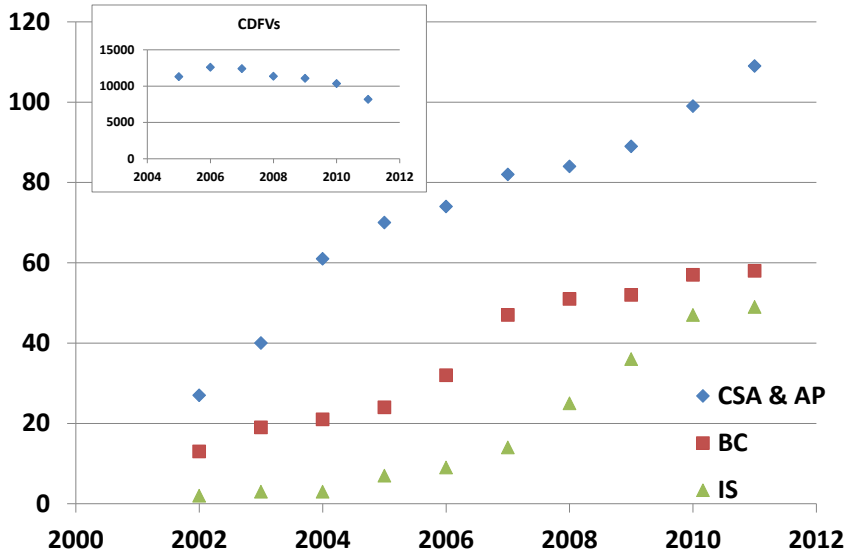


**Figure 5.** The number of States with CSA & AP in force, the number of States with broader conclusion and IS implemented is plotted. On the subgraph, the IAEA's in-field effort is also plotted in CDFV unit for the States with CSA & AP in force.

It can be seen, that while the number of States with CSA & AP has increased steadily since 2007, by introducing IS for higher portion of the States with broader conclusion (BC), the IAEA could reduce the overall CDFVs, thus achieving higher efficiency. A significant decrease in 2011 (21%) in CDFV illustrates clearly the Agency's success in redistributing resources from in-field activities to analysis of all safeguards information on the State-level, while maintaining its ability to draw independent conclusions. This is however only possible by introducing enhanced remote monitoring measures and making more use of the SSAC/RSAC effectiveness and co-operation capacity.

## 5. Overall Safeguards efforts

In the previous chapters it could be shown that the introduction of IS in a State the on-site efforts of the IAEA have decreased noticeable. The amount of reduced activities depends mainly on the type of facilities, e.g. most of the reduction it is seen at LWRs. In other fields, like for spent fuel transports the IAEA IS concept is still not (well) developed. Currently no or very little reduction is observed for such activities.

Whereas the on-site IAEA inspections have decreased after IS implementation, the scrutiny of the quality and completeness of the States' reporting (mostly nuclear material accounting and AP-declations) is more in depth. IAEA use of open sources has increased, but from States' experience it is not clear which selections criteria are used to choose and evaluate the latter. This in depth information collection and evaluation had led, especially during the phase before the drawing of broader conclusion, to a quite big amount of questions and clarifications to be provided by the State. Although the peak of this phase after the broader conclusion has decreased, it is still at a higher level than during the traditional Safeguards era pre-AP. In fact the broader conclusion for a State is a continuous process and it is drawn at an annual basis. Therefore, it is expected that clarifications and answers to the IAEA will remain at a certain level. The fact that these are often concerning the broader aspects of the nuclear fuel cycle and less at a facility level, the State's effort to gather the required information is quite high since not always they are available at hand.

In conclusion at one hand the on-site activities are reduced but on the other hand headquarters' activities, for both SSAC/Euratom and IAEA, are taking more efforts.

## 6. Role of SSAC/Euratom in IS implementation

During the process of the integrated safeguards implementation in all EU States, the IAEA streamlined and enhanced its safeguards activities through the increased use of safeguards schemes such as

SNRIs with mailbox provision of operational data of the relevant facilities, remote monitoring of safeguards data generated by installed C/S systems or measurement devices and complementary access activities. The implementation of random inspections and complementary access requires however much more flexibility on the part of the facility operators, State authorities and Euratom, regarding the scheduling of inspections and the provision of inspectors with prompt access to nuclear sites and other locations.

Both the national authorities as well as the Euratom system of safeguards at EU level have specific possibilities in establishing the necessary flexibility in their own operations and in those of the facilities. While some SSAC have a greater flexibility and influence on the facilities through various national authorization instruments if compared to regional systems, Euratom has been able to negotiate some elements of the IAEA IS implementation through the new partnership approach mechanism. During the initial discussions on the practical implementation aspects of IS in the EU in 2007-2008, Euratom had made it clear from the beginning that it has its specific own safeguards responsibilities under the Euratom Treaty and under the verification agreement INFCIRC/193. It was therefore especially important that approaches could be found which would allow both the IAEA and EURATOM to fulfil their respective obligations under INFCIRC/193 and INFCIRC/193/Add.8 and which are adapted to reflect the IAEA's integrated safeguards and the European Commission's strategy for improved safeguards effectiveness. This was done through establishing so-called Partnership Approaches (PA) for all major types of facilities (light water reactors, spent fuel storages, research reactors, gas centrifuge enrichment plants and fuel fabrication plants). In addition, facility specific partnership approaches were developed for the major bulk handling facilities and selected other facilities. Through the use of surveillance or other technical means it was possible to reduce the number of facilities where unannounced inspections are carried by the IAEA alone (less then 10 facilities in the whole EU) and use random inspections with advance notice instead. This allows Euratom to make the maximum use of random inspections for the attainment of Euratom inspection goals. It also allowed the IAEA and Euratom to continue with their established schemes of common use of instruments, sharing of data and sharing of costs. Moreover, the common obligation of keeping the safeguards related burden on operators at a minimum could be preserved.

A rolling scheme of inspections was developed that envisages the regular provision by the IAEA to the Commission of advance information of the random interim inspections (RIIs) with short notification required under Integrated Safeguards approaches in the EU. The early provision of this information gives the Commission the opportunity to plan the resources required and to make arrangements for full participation in such inspections. This information would however not identify the facility or State concerned, but merely the number of inspections planned in a particular time envelope in the future, thereby keeping randomness and unpredictability of IAEA inspections. Annually, about 50 – 60 inspections of this kind are carried out in the EU.

A number of PA support documents was also developed to address such topics as joint-use remote data transmission, common use surveillance, common use electronic seals etc.

The further development of the IAEA state level concept combined with the unique nature of the Euratom system of safeguards as a system comprising 27 European States still provides more potential for increased efficiency of safeguards implementation in the EU. This then will also help to increase safeguards effectiveness on a world-wide scale.

## 7. Conclusions

IAEA in-field PDI effort has changed depending on facility types/fuel cycle of the State. Reduction is justified by the broader conclusion for a State and also by safeguards technology development. The redistribution of resources is used for more State level analysis work or to other areas of the world with higher proliferation risks than the EU.

Euratom in-field safeguards efforts also reduced in some States due to technological development in safeguards (extensive use of RDT) and revised common approaches with the IAEA resulting in reduced inspection frequencies for spent fuel verification in reactors and spent fuel storage facilities.

Due to the greater flexibility required by the unpredictability elements of the State level IS approach increased effort have to be allocated from the SSAC/operator side. This could even be increased in

cases where the IAEA"s more intensive use of State level analysis (open source) triggers more communication and requests for information and clarification/confirmation from SSAC/RSACs.

There is a unique role of a politically acknowledged regional safeguards system in the finalization of facility IS approaches that meets also the regional safeguards requirements, which could be deployed more intensively to reduce IAEA in-field verification activities.

One of the possible way for the IAEA to sustain the effective global safeguards system with limited resources is making greater use of SSACs/RSACs with IS implemented. The IAEA seems to be on the right track with current developments on State level safeguards approaches.

## 8. Acknowledgement

The IS WG chair acknowledges all the States that voluntarily participated in this review and the efforts of all the IS WG members in developing the state information sheets. Special thanks should go to Wolfgang Kahnmeyer, EC, DG ENER-E4 and Christos KOUTSOYANNOPOULOS, EC, DG ENER-E1 for their valuable contribution.

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# Development of new analytical capabilities for nuclear forensics at CEA/DIF

**Fabien Pointurier, Amélie Hubert, Anne Salaün, Mickaël Mendes, Céline Rodriguez, Anne-Claire Pottin, Anne-Laure Fauré, Olivier Marie, Jean Aupiais**

CEA, DAM, DIF, F-91297 Arpajon, France
E-mail: fabien.pointurier@cea.fr

***Abstract:***

*Analytical laboratories from CEA/DIF, a CEA centre located 35 km South of Paris, are part of the NWAL (Network of Analytical Laboratories in support of IAEA's nuclear safeguards) for the analysis of environmental samples since 2001 for both bulk and particle analysis. In addition to the knowledge established more than 15 years ago for analysis of environmental samples and continuously improved since then, new analytical capabilities applicable to nuclear forensics expertises are currently under development. The two main projects are the "age" determination of uranium materials (so-called "datation", i.e. measurement of the date of the last chemical purification applied to the material) and "geo-location" of uranium-ore concentrates (i.e. determination of the type of uranium-mine and/or the geographical origin of the material).*
*As only limited amounts of nuclear materials can be admitted in CEA/DIF facilities normally dedicated to trace analysis of environmental samples, methodologies are developed with a view to be applicable to as low as possible quantities of uranium. Then, age determination procedures require only micrograms or milligrams of uranium, when analyses are respectively carried out through measurements of the $^{230}Th/^{234}U$ or of the $^{231}Pa/^{235}U$ ratios. Regarding the geo-location, we developed three different methodologies to determine respectively the isotopic compositions of uranium ($^{235}U/^{238}U$ and $^{234}U/^{238}U$), oxygen ($^{18}O/^{16}O$), and relative concentrations of rare earth elements (REE or lanthanides) from very small amounts of uranium-ore concentrates.*
*In this paper, we describe the analytical procedures for uranium age determination and for REE measurements and give a few examples of application on standard reference materials.*

**Keywords:** nuclear forensics, uranium, uranium-ore concentrates, age determination, datation, geo-location, rare earth elements, lanthanides, standard reference materials.

## 1. Introduction

As part of the nuclear proliferation control program, the CEA/DAM — Ile de France (CEA/DIF) trace analysis laboratories at Bruyères-le-Châtel regularly perform isotopic analyses on swipe samples taken by IAEA (International Atomic Energy Agency, Vienna, Austria) inspectors in facilities in order to collect particles that separate from nuclear materials during the various treatments applied to them [1]. One other important item of information is the date of production or last purification of the material [2]. This information is valuable both in non-proliferation, to verify the declarations of nuclear facility operators or detect undeclared activities, and in the field of nuclear forensics, the goal of which is to identify the use, origin, routing, and operational history of seized nuclear material. The date of purification of the material can be obtained by determining the ratio between one of the isotopes of the element of interest and its radioactive decay product. During chemical purification of uranium, the radioactive daughter products of uranium and the impurities in the composition of the material are eliminated. The daughter isotopes $^{230}Th$ and $^{231}Pa$, direct decay products of the parent isotopes $^{234}U$ and $^{235}U$, subsequently increase from the purification date at a rate depending on the radioactive decay rate of the parent isotope (radioactive half-lives of $2.45\times10^5$ years for $^{234}U$ and $7.04\times10^8$ years for $^{235}U$). Several laboratories have developed analytical methods (chemical purification and isotopic measurement, usually by mass spectrometry and/or a spectrometry) for determining the $^{230}Th/^{234}U$ ratio [3-8] or the $^{231}Pa/^{235}U$ ratio [9,10]. However, these determinations require at least 100 µg, and in practice several milligrams, of uranium. Age determination through measurement of the $^{231}Pa/^{235}U$ ratio is more difficult because of the complicated chemistry of protactinium. The

approach undertaken at the CEA/DIF is the development of uranium age determination procedures through measurement of both isotope ratios. These methodologies must be useable even when the seized or sampled masses are very limited, practically in the range of environmental samples. However, the main difficulty of these analyses on such small quantities of material is that the masses of the daughter isotopes, $^{230}$Th and $^{231}$Pa, may be extremely low, of the order of a few femtograms (1 fg = $10^{-15}$ g).

Another indicator of great interest in the field of nuclear forensics is rare earth element (REE) content in uranium-ore concentrate (UOC) samples. As a matter of facts, relative REE concentrations are reportedly linked to the geological type of uranium deposit [11-17]. Even through absolute concentrations are probably variable from one part of the U-deposit to the other and surely vary throughout the industrial treatment applied to the U-ore to finally obtain UOC, the original distribution of REE concentrations is reportedly unchanged. However, the main difficulty is that, after chemical purification, the REE concentrations are very low, at trace level, with concentrations probably ranging from the ppb (ng/g) to the ppm (µg/g). However, a few papers described REE measurements in UOC, either directly without REE separation from the uranium matrix, or by extracted REE and measuring them in purified solutions. Again, originality of our approach is to develop an analytical procedure applicable to relatively low amounts of uranium, typically in such a case in the milligram range. As the instrument used for the elemental measurement is an environmental one, usually dedicated to measurement of minute amounts of uranium in environmental samples, most of the uranium matrix must beforehand be eliminated through a chemical purification.

In this paper, we describe the analytical procedures developed in the laboratory for age determination of uranium oxides and for measurement of REEs in UOC samples. Lastly, we describe the application of the developed analysis procedure to small quantities of uranium from standard reference materials in order to validate the methods.

## 2. Uranium age determination through measurement of the $^{230}$Th/$^{234}$U ratio

As mentioned before, the aim of this study was to develop a method for “dating” small quantities of U (less than one µg) through the measurement of the $^{230}$Th/$^{234}$U ratio. This method was described with more details in a previous paper [18].

### 2.1. Uranium – thorium purification

The requirements for the U/Th purification protocol are very strict. At first, the analytical procedures must have excellent thorium extraction efficiencies from a uranium matrix and be very robust with regard to risks of thorium contaminations from reagents or the atmosphere. A "microchemistry" technique using extremely small quantities of reagents is thus ideal. The volume of the chromatographic separation resin was therefore reduced to 100 µl, compared with several ml for the separations customarily performed by the laboratory on environmental samples. Purification columns in small funnels were prepared for this purpose, and the quantities of reagents used were substantially reduced. Purification times were also much shorter, greatly reducing exposure times to the atmosphere. Under these conditions, thorium contamination was very low, less than 2 pg of $^{232}$Th and less than one fg of $^{230}$Th. Lastly, much smaller quantities of liquid waste were generated. The purification protocol is given in Figure 1.

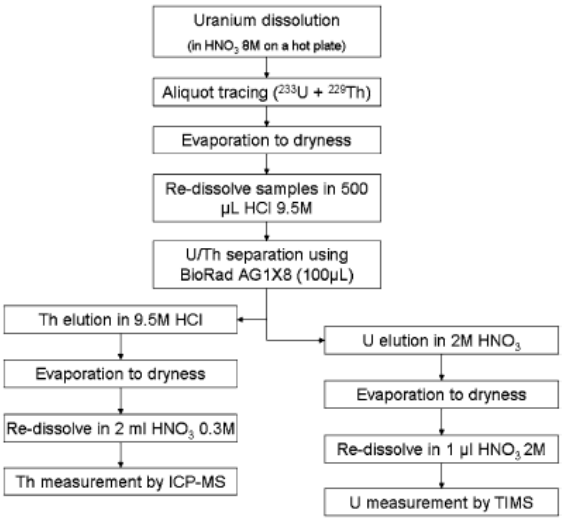


**Figure 1:** Chemical treatment protocol for uranium dating by measurement of the $^{230}$Th/$^{234}$U ratio.

### 2.2. Uranium and thorium isotopic measurements

The isotopic measurement technique for thorium must have excellent measurement sensitivity, with detection limits of the order of femtograms. For this, the CEA/DIF has a double-focusing inductively coupled plasma mass spectrometer (ICP-MS), used routinely for ultra-trace measurements of plutonium in

environmental samples [19,20], which has the required performance.
The isotopic composition of uranium is determined using a "Triton" TIMS (Thermo-Fischer Scientific, Bremen, Germany). One advantage of TIMS is that the risks of instrumental contamination are very limited as consequence of its operating principle. After chemical purification, the samples were deposited on a previously degassed double filament made of high-purity rhenium. The filaments were inserted into the ionization source and heated gradually until a significant signal was obtained. The analysis was performed in dynamic multi-collection mode. The ion intensities corresponding to isotopes $^{233}$U and $^{234}$U were measured with an electron multiplier and those of isotopes $^{235}$U and $^{238}$U with Faraday cups. Uranium and thorium were quantified using the isotope dilution technique, adding known quantities of isotopes $^{233}$U (IRMM, Geel, Belgium) for uranium and $^{229}$Th (AEA Technology, Harwell, UK) for thorium to the samples.

### 2.3. Validation of the U/Th age determination method

The method was validated with µg amounts of the standard reference material NBS U010. Process blanks contain only 1-2 pg of $^{232}$Th and less than one fg of $^{230}$Th. Moreover, the chemical yield for thorium is high (~94% on average) and reproducible. Lastly, correct agreement with the date of certification was obtained (see Figure 2), which is encouraging if we consider that the date of certification was probably close to the date of chemical purification applied to the material. Other analyses are currently in progress with other CRM.

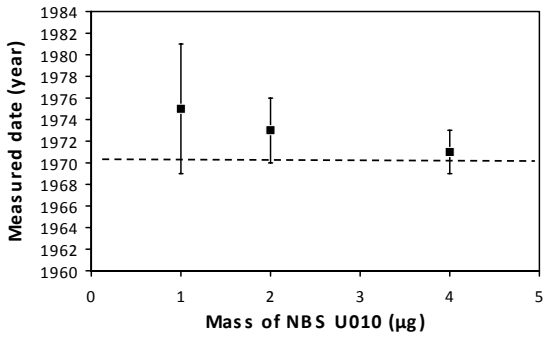


**Figure 2:** Production dates of the uranium specimen determined by analysis of 1, 2, and 4 µg of the material. The dashed line corresponds to the date reported on the standard's certificate. Uncertainties are standard uncertainties (coverage factor of 1).

## 3. Uranium age determination through measurement of the $^{231}$Pa/$^{235}$U ratio

The aim of this study was to develop a method for “dating” relatively small quantities of U, in the µg to the mg range, through the measurement of the $^{231}$Pa/$^{235}$U ratio.

### 3.1. Uranium – protactinium purification

Protactinium chemistry is very complex and tricky. Protactinium notably tends to be adsorbed on glass surfaces. To define an efficient purification procedure, we carried out a new determination of distribution coefficients ($K_d$) for U, Pa, Th on Dowex AG1X8 resin. This work is currently under publication.
One other difficulty of uranium age determination by measurement the $^{231}$Pa/$^{235}$U ratio, is that the only other Pa isotope usable for isotope dilution mass spectrometry (IDMS) is $^{233}$Pa, which is a short-life isotope ($T_{½}$ = 24.7 days). Therefore, the $^{233}$Pa tracer must be produced from a $^{237}$Np source only a few days before the analysis. Then, a first chemical purification is carried out to extract a sufficient amount of $^{233}$Pa from the $^{237}$Np solution. To optimize such purification, the distribution coefficients for Pa and Np for the Dowex AG1X8 were determined precisely. Uranium was also quantified using the isotope dilution technique, adding known quantities of $^{233}$U.
Starting from the uranium sample, Pa, U and Th separations from the uranium matrix are carried out with the same AG1X8 resin. Then, Pa and Th are further purified from uranium using other AG1X8 columns. $^{233}$Pa activities are measured by means of gamma-spectrometry before the first purification columns and after the last Pa purification column, to determine the Pa chemical yield. Therefore, this method allows performing uranium age determination using both $^{230}$Th/$^{234}$U and $^{231}$Pa/$^{235}$U ratios. However, ages calculated with the $^{230}$Th/$^{234}$U are presently almost systematically overestimated with regard to the ones given by the $^{231}$Pa/$^{235}$U ratios. This discrepancy is not yet explained.

### 3.2. Uranium and protactinium isotopic measurements

Uranium and protactinium isotopes are measured either using the multi-collector TIMS Thermo "Triton" or a double focusing ICP-MS Thermo "Element XR". This instrument is equipped with the so-called "Jet interface" which strongly enhance the instrument's

sensitivity, enabling absolute detection limits lower than one fg.

### 2.3. Validation of the U/Pa age determination method

Various analysis carried out on several isotopic CRMs (NBS U010, U030, U100, U500, U950) showed that the chemical yield for Pa is high enough (>70%) and that the Pa purification factor from U is satisfying (~$2\times10^6$). Besides, $^{231}$Pa content in process blanks is below the instrument detection limit, i.e. below one fg.
Unfortunately, we do not know the date of purification of the analyzed CRMs available in our laboratory, with exception of the U100, which had been purified in a single batch the 8 January 1959 [8]. Our first results show an acceptable agreement with this date, even if the corresponding age is slightly overestimated (see Figure 3). There is a lack of CRM suitable for age determination (i.e. with "certified" purification dates).

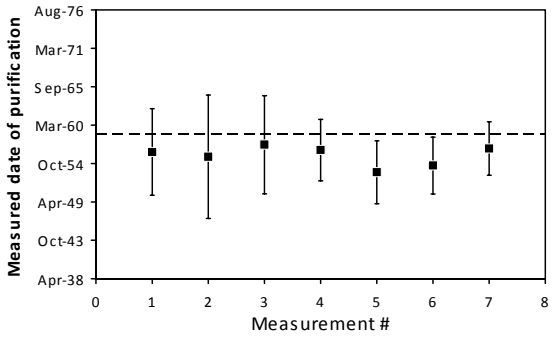


**Figure 3**: results obtained for age determination of small amounts of the NBS U100 CRM. Analyzed masses ranged from 4 µg to 600 µg. The dashed line corresponds to the reported purification date of the single batch produced [8]. Uncertainties are expanded uncertainties, with a coverage factor of 2.

## 4. Lanthanide measurement in UOC

### 4.1. Analytical methodology

As mentioned before, most of the uranium matrix must be eliminated to avoid injection of a too high uranium concentration into an ICP-MS dedicated to trace analysis of actinides. Therefore, a thorough chemical purification must be carried out. This is achieved thanks to a 4 ml – column of chromatography resin. As no UOC CRM certified for all REEs is available; REE quantification is carried out thanks to a calibration curve. For each analysis, three determinations are made from one single aliquot of 20-100 mg of UOC (powder).

### 4.2. Validation of the REE measurement procedure

The developed analytical procedure was validated thanks to both known reference REE solution (Spex "CLMS-1") and with two $U_3O_8$ reference materials ("Bolet" and "Morille", CEA / CETAMA, Marcoule, France) for which concentrations are certitfied for 4 REEs (Sm, Gd, Eu, Dy). Our results and the reference and published data [17] are in very good agreement for all of these four elements (see Figure 4 for dysprosium).
These analyses also showed that the remaining uranium quantity is no significant and do not produce a measurable memory effect in our instrument. Moreover, the chemical yield is close to 100%. Other tests were carried out on other CRMs and on a few UOC of interest. A paper gathering these results and more details about the analytical procedure will be published soon.

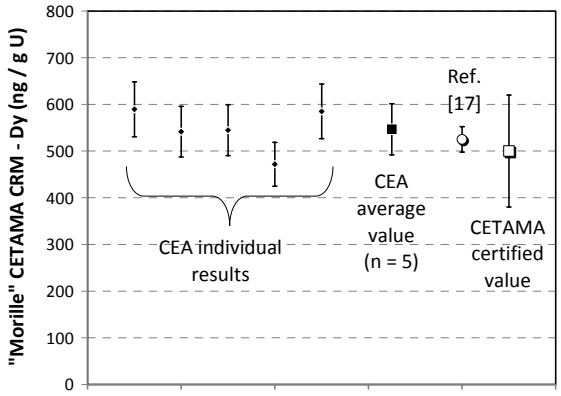


**Figure 4:** comparison of results obtained in our laboratory for the measurement of dysprosium concentration in the "Morille" CRM, with published [17] and certified concentrations.

## 4. Conclusions

Operational methods are now available at CEA/DIF for uranium age determinations. These methods are applicable to "low amounts" of uranium, typically in the µg – mg range. An operational method is also now available for REE measurement in UOC. By the same way, this analytical procedure can be applied to limited amounts of uranium, in the mg range. Other developments are currently under progress for measurement of other potentially relevant indicators for nuclear forensics, namely oxygen isotope ratio for determination of the geographical origin of the oxygen atoms present in uranium oxides, and uranium chemical composition (i.e. stoechiometry) by means of Raman spectrometry. Both

measurement techniques will be applicable to micrometer-size particles. Now, CEA/DIF analysts need to gain experience by applying the newly developed procedures on various samples. Nevertheless, for validation of all of those analytical methods, there is a strong need of various CRMs for which relevant characteristics are certified (date of purification and level of purification for age determination, REE content at trace level, oxygen isotopic composition, etc.).
Lastly, refurbishing and extending of a dedicated "hot" laboratory (controlled area) are in progress at CEA/DIF, to expand capabilities towards higher amounts of nuclear material. Expected completion of this new facility is beginning 2014.

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# Particle Size Inhomogeneity Effect on Neutron Resonance Densitometry

**B. Becker[1], H. Harada[2], K. Kauwenberghs[1], F. Kitatani[2], M. Koizumi[2], S. Kopecky[1], A. Moens[1], P. Schillebeeckx[1], G. Sibbens[1], H. Tsuchiya[2]**

[1]EC-JRC-IRMM
Unit D.4. Standards for Nuclear Safety, Security and Safeguards
Retieseweg 111, B-2440 Geel, Belgium
E-mail: bjorn.becker@ec.europa.eu, peter.schillebeeckx@ec.europa.eu

[2]Japan Atomic Energy Agency (JAEA)
Tokai-mura, Naka-gun, Ibaraki 319-1195, Japan
E-mail: harada.hideo@jaea.go.jp

***Abstract:***

*Neutron Resonance Densitometry (NRD) represents a possible option to determine the heavy metal content in melted nuclear fuel. This method is based on the well-established methodology of neutron time-of-flight (TOF) transmission and capture measurements. In particular, NRD can measure both the isotopic and the elemental composition. It is a non-destructive method and is applicable for highly radioactive material. The details of this method are explained in another contribution to this symposium.*

*The accuracy of NRD depends among other factors on sample characteristics. Inhomogeneities such as density variations in powder samples can introduce a significant bias in the determination of the composition. In this contribution, the impact of the particle size distribution of such powder samples on results obtained with NRD is investigated. Various analytical models, describing the neutron transport through powder, are compared. Stochastic numerical simulations are used to select a specific model and to estimate the introduced model uncertainty. The results from these simulations will be verified by dedicated measurements at the TOF-facility GELINA of the EC-JRC-IRMM.*

**Keywords:** non-destructive assay; densitometry; neutron time-of-flight; resonance analysis; melted fuel; severe accidents; nuclear safeguards

## 1. Introduction

Neutron Resonance Densitometry (NRD) can be used to determine the isotopic and the elemental composition of unknown samples [1,2]. NRD consists of a combination of Neutron Resonance Transmission Analysis (NRTA) [3,4,5,6] and Neutron Resonance Capture Analysis (NRCA) [4,5]. Both NRTA and NRCA are non-destructive neutron time-of-flight measurement techniques which are based on well-established methods for nuclear cross section determination [7].

In case that the sample consists of particle-like debris of melted nuclear fuel formed in a severe accident, such as happened in the Fukushima Daiichi nuclear power plants, the inhomogeneity of the sample can introduce a significant bias in the result of an NRD measurement. In this contribution, the impact of the sample inhomogeneity is investigated by comparing different analytical methods with stochastic simulations of the neutron transport through powder samples.

## 2. Particle Size Models

In the last decades various models have been developed to describe the neutron transport through stochastic media [8]. Here, we consider binary stochastic mixtures of particles (phase $\alpha$) embedded in a matrix (phase $\beta$). The mixture is characterized by the volume fraction of each phase $i = \alpha, \beta$ defined as $p_i = V_i/(V_\alpha + V_\beta)$

where $V_i$ is the volume, and by the chord lengths $\lambda_i = 4V_i/S$ where $S$ is the boundary surface between the different phases. The volume fraction of the particles is also referred to as packing fraction.

In the following, a limited number of models available to calculate the neutron transport through a stochastic mixture are discussed.

## 2.1. Homogeneous Limit

The homogeneous limit, also called atomic mixing, is based on the assumption that the neutron effectively sees a mixture of particles and matrix material. This is the case if the neutron passes multiple particles between interactions, i.e. if the mean free path ($mfp$) of the neutron in the particle is significantly higher than the characteristic particle size $l$: $mfp = 1/\Sigma^{\alpha} \gg l$. In this case the macroscopic total cross section of the particle-matrix mixture $\Sigma^{hom}$ is simply given by the volume fraction weighted sum of the particle and matrix total cross sections $\Sigma^{\alpha}$ and $\Sigma^{\beta}$ :

$$\Sigma^{hom} = \bar{\Sigma} = p_{\alpha}\,\Sigma^{\alpha} + p_{\beta}\Sigma^{\beta}.$$

## 2.2. Macroscopic Model

Kopecky et al. [9] developed a macroscopic model to describe the transmission of neutrons through a Pu powder sample. The model is based on the assumption that the probability of a neutron passing a particle is given by Poisson statistic and that this probability results in a log-normal distribution of the sample thickness with an additional hole fraction $f_h$. The transmission (T) through the sample is then given by:

$$\mathrm{T} = (1 - f_h)\int e^{-n'\sigma^{\alpha}x}\,p(x)dx + f_h$$

where $n'$ is related to the areal density $n$, determined from measurements of the mass and sample area, by: $n' = n/(1 - f_h)$. $\sigma^{\alpha}$ is the total cross section of the particle. The variable $x$ is distributed as a log-normal distribution:

$$\mathrm{p}(x) = \frac{1}{x\sqrt{2\pi s^2}}exp\left(-\frac{(\ln x + s^2/2)^2}{2s^2}\right)$$

with an average value one and the width parameter $s^2$. It is assumed that the cross section of the matrix material can be neglected. The width parameter and hole fraction are free model parameters. In general, Kopecky's model is not restricted to a log-normal distribution of the thickness. Depending on the powder characteristics, a different thickness distribution might be advantageous. The model of Kopecky et al. has been implemented in the least squares fitting program REFIT [10].

## 2.3. Microscopic Models

### 2.3.1. Analytical Expressions

- **Burrus's Model**: Burrus [11,12] investigated the transmission of thermal neutrons through boral slabs made out of a heterogeneous mixture of boron carbide and aluminium. Burrus developed a simple model for the effective total cross section $\Sigma_{eff}$ of the mixture neglecting the matrix cross section. The effective cross section is given by:

  $$\Sigma^{B}_{eff} = -\frac{1}{\lambda_{\alpha}}ln(1 - p_{\alpha}\,F)$$

  where $F$ is the probability that a neutron does not penetrate a particle. $F$ is related to the self-collision probability $P_c$ by $F = \lambda_{\alpha}\Sigma^{\alpha}(1 - P_c)$. The self-collision probability can be determined using Wigner's rational approximation:

  $$P_c = \lambda_{\alpha}\Sigma^{\alpha}/(1 + \lambda_{\alpha}\Sigma^{\alpha}).$$

- **Doub's Model**: Doub's model [13] is based on calculating a self-shielding factor $f_D$ for the particles. The average transmission through a mixture of thickness $R$ is then given by $\bar{T} = \exp(-R\,p_{\alpha}f_D\,\Sigma^{\alpha})$. Doub assumed a binominal distribution of the number of particles that the neutron passes. The self-shielding factor of mono-dispersed spheres with radius $r$ is:

  $$f_D = \frac{1}{\frac{2}{3}y\,{}^{p_{\alpha}}/_{g}}ln\left(\frac{1}{1 - {}^{p_{\alpha}}/_{g}\,(1 - \bar{t})}\right)$$

  with $y = 2r\Sigma_{\alpha}$ and $g = 0.740$. $\bar{t}$ is given by $\bar{t} = 2/y^2\ [1 - (1 + y)e^{-y}]$. Doub extended the model to polydisperse spheres using a volume averaged self-shielding factor $\bar{f}_D = \sum_j f_{D_j}V_j/V_{\alpha}$ where $f_{D_j}$ and $V_j$ are the self-shielding factor and volume of spheres with a radius $r_j$. Doub compared his model to measurements of transmission through a mixture of boron-carbide and aluminium spheres [13].

- **Randall's Model**: While Doub's model is restricted to spherical particles, Randall developed a more general model based on

a stochastic description of the integral transport equation (for more details see Refs. [12,14,15]). When all particles have the same characteristics, the average transmission is given by:

$$\bar{T} = \left\{p_\beta + p_\alpha\, E'\right\}^N e^{-R\,\Sigma^\beta}$$

with $E' = [1 + \lambda_\alpha \Sigma(\sigma_\Delta/\lambda_\alpha)]^{-[\lambda_\alpha/\sigma_\Delta]}$. The particle shape is describe by the average chord length $\lambda_\alpha$ and the chord length variance ${\sigma_\Delta}^2$. The factor $N$ is given by $N = (R\ a)/\lambda_\alpha$ with the effective linear packing factor $a \cong [(2 - pf)X]^{-1}$ where $X = 1 + {\sigma_\Delta}^2/\bar{l}^2$. Values for $\lambda_\alpha$ and $X$ for different particle shapes are tabulated in Ref. [14]. Randall further extended the description of the transport based on binominal statistics [15]. However, since the final expression involves the calculation of the confluent hypergeometric function, the calculation tends to be slow and sometimes not stable. Therefore Randall's extended expression is not considered in this contribution.

- **LP Model**: Levermore et al. [16] and Vanderhaegen [17] developed a model of the neutron transport through a statistical mixture for which the transition from one phase to the other is described by a Markov process. For a stationary process the transmission through a sample with thickness $R$ is given by:

$$T = \left\{\frac{r_+ - \tilde{\Sigma}}{r_+ - r_-}\right\} e^{-r_+ R} + \left\{\frac{\tilde{\Sigma} - r_-}{r_+ - r_-}\right\} e^{-r_- R}$$

with $\tilde{\Sigma} = p_\beta \lambda_\alpha + p_\alpha \lambda_\beta + {\lambda_\alpha}^{-1} + {\lambda_\beta}^{-1}$ and $2r_\pm = \bar{\Sigma} + \tilde{\Sigma} \pm \sqrt{\left(\bar{\Sigma} - \tilde{\Sigma}\right)^2 + 4\theta}$. The parameter $\theta$ is given by: $\theta = \left(\Sigma^\alpha - \Sigma^\beta\right)^2 p_\alpha p_\beta$.

### 2.3.2. Monte Carlo Simulations

The transmission and capture of neutrons in stochastic media can also be calculated by Monte Carlo (MC) simulations. The idea is to generate multiple stochastic geometries and to track individual neutrons through these geometries. Based on a significant number of histories, the average transmission can be deduced. In this contribution, we explore two methods to generate the stochastic geometry:

- **MC Method 1**: Spheres are randomly placed into a control volume until a specific packing fraction is obtained similar to the Random Sequential Addition (RSA) algorithm described in Refs. [18,19]. Newly placed spheres are not allowed to overlap with already placed spheres. In case of polydisperse spheres, spheres with a large radius are placed first. A limitation of this method is that the maximum reachable packing fraction is lower than the theoretical one of a closely packed mixture. Figure 1 shows a cut through a polydisperes mixture.

- **MC Method 2**: The algorithm of Switzer [19] is used to generate a 2-dimensional geometry with Markovian characteristics (for details see Refs. [20,21]). Following the method used by Tibault et al. [21] a binary mixture is generated. Figure 2 shows an example of a binary stochastic mixture.

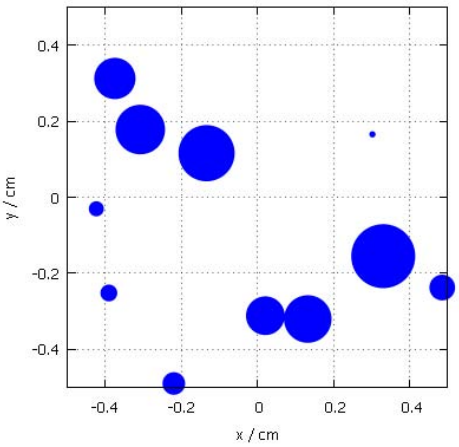


**Figure 1:** Example of a stochastic geometry based on a polydisperse spherical particle distribution.

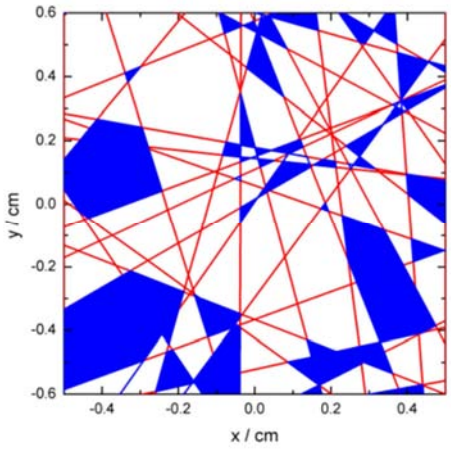


**Figure 2:** Example of a stochastic Markovian geometry.

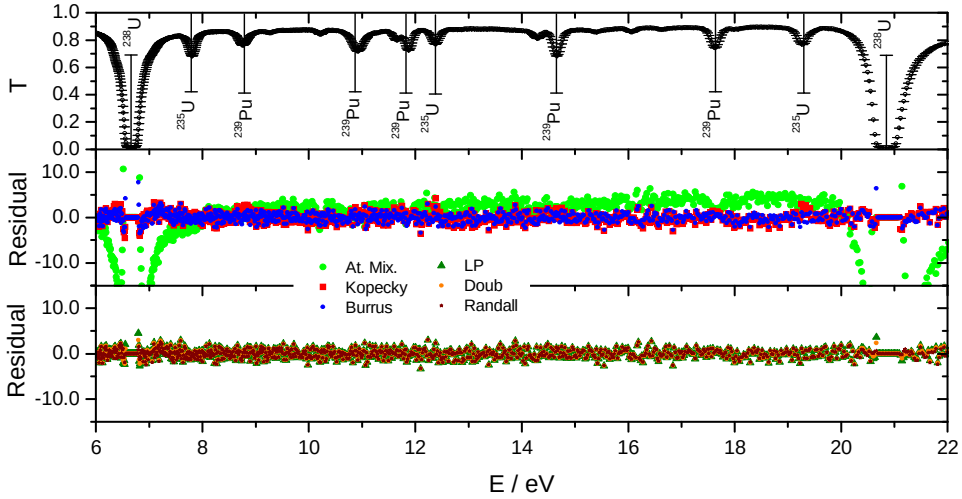


**Figure 3:** Created transmission spectrum and residual distribution in benchmark case 1.

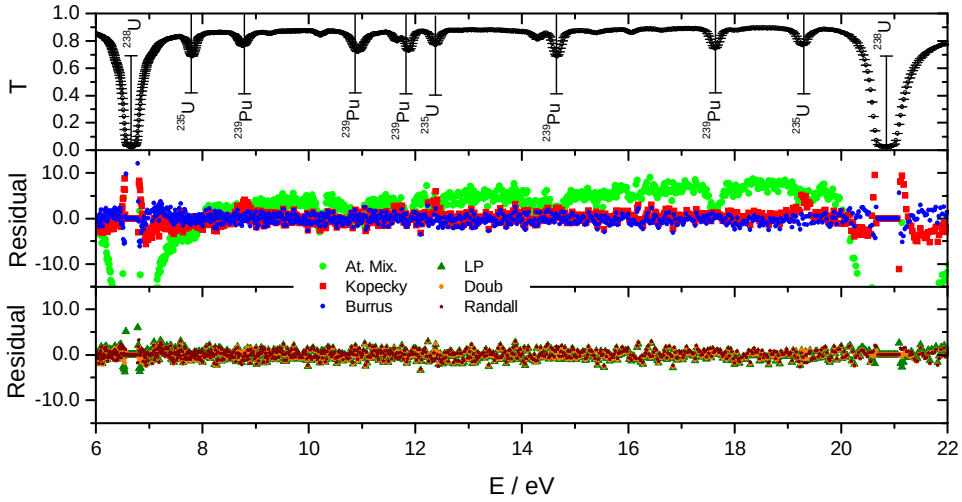


**Figure 4:** Created transmission spectrum and residual distribution in benchmark case 2.

## 3. Particle Size Impact on NRTA

A bias introduced by the particle size distribution on results of an NRTA analysis was investigated by comparing analytical model calculations with MC simulations. The two MC methods described in section 2.3.2 are used to generate a set of synthetic transmission data. In all cases the particle phase consists of uranium and plutonium oxide with a density of 10.97 g/cm$^3$. The particle composition is given in Table 1. The matrix is considered to be void. The packing fraction was assumed to be 30%. The sample thickness was chosen to be 1 cm.

The six different analytical methods given in sections 2.1, 2.2 and 2.3.1 were used to determine by $\chi^2$ minimization the average areal density of the $U^{235}$, $U^{238}$ and $Pu^{239}$ content. In case that the transmission is less than 0.1, the data points were not included in the adjustment. All free model parameters (e.g. hole fraction, average diameter or packing fraction) are adjusted on the same footing. An additional constant cross section has been adjusted in order to take into account the contribution of the oxygen cross section. This is equivalent to adjusting the normalization of the transmission.

| Nuclei | at.% |
|---|---|
| $U^{235}$ | 0.6667 |
| $U^{238}$ | 31.999 |
| $Pu^{239}$ | 0.6667 |
| $O^{16}$ | 66.667 |

**Table 1:** Assumed particle composition.

### 3.1. Polydisperse Spheres

Two benchmark cases were generated using MC method 1. A flat particle diameter distribution ($0\,mm < d < 2\,mm$) and a log-normal distribution (with: $\bar{d} = 1\,mm$ and the

width parameter: $\sigma = 0.6$) were used for benchmark cases 1 and 2, respectively. Sampled particles were placed in a volume with a thickness of 1 cm and a height and width of 4 cm. About 4600 spheres were sampled to reach the desired packing fraction of 30%. The created geometry was used to evaluate the thickness of the particle path along 40000 different transmission paths. From the obtained thickness distribution a transmission spectrum was created. A 1% uncertainty for transmission $T = 1$ has been associated to the spectrum. The final transmission spectrum was then obtained by adding a Gaussian fluctuation to the spectrum corresponding to the uncertainty.

In case of Burrus's model all parameters were found to be strongly correlated, therefore an initial uncertainty of the packing fraction of 10% was assumed. In case of Randall's model, the parameter $X$ was chosen to be 9/8 which is in principle only correct for monodisperse spheres. Figures 3 and 4 show the created transmission spectrum for cases 1 and 2, respectively, as well as the residual of the parameter adjustment using the different methods. By simply homogenizing particles and matrix, a significant bias is introduced when adjusting the average areal densities. For both cases the bias is up to 5% for $^{235}$U and $^{239}$Pu and 13% for $^{238}$U. The other models perform more or less equally. They all improve the result of the adjustment significantly compared to homogenization.

### 3.2. Markovian Geometry

Two additional benchmark cases (case 3 and 4) were created using the second MC method. A mean chord length ($\lambda_A$) of 0.067 cm and 0.267 cm were assumed for cases 3 and 4, respectively. 400 different realizations of the stochastic geometry were sampled for each case. 4000 neutron histories per stochastic geometry were simulated. Capture and scattering events were taken into account by neutron weight reduction. Figure 5 and 6 show the two created transmission spectra. The transmission deviates in particular in the region close to the $^{238}$U resonances. The uncertainties due to sampling range from 0.002 to 0.006 depending on the transmission. Again, the initial uncertainty of the packing fraction of 10% was assumed for Burrus's model and the parameter $X$ in Randall's model was assumed to be 9/8.

Theoretically, the LP model should describe perfectly the transmission through a stochastic geometry created by MC method 2. This is indeed observed for both cases. The nominal densities and the adjusted densities using the LP model agree within the uncertainty of the fit. For case 3, Doub's and Randall's model give relatively good results with a bias of less than 1.5%. However, the bias increases for case 4 due to the increased particle size. Atomic mixing always leads to an under prediction of the areal densities. For case 3, the bias is up to 6% for $^{235}$U and $^{239}$Pu and 12% for $^{238}$U. For case 4, this bias increases up to 20% for $^{235}$U and $^{239}$Pu and to 35% for $^{238}$U. Even though the fitting residual distribution of Kopecky's or Burrus's model is more or less flat, both models over predict the densities by up to 25% for case 4.

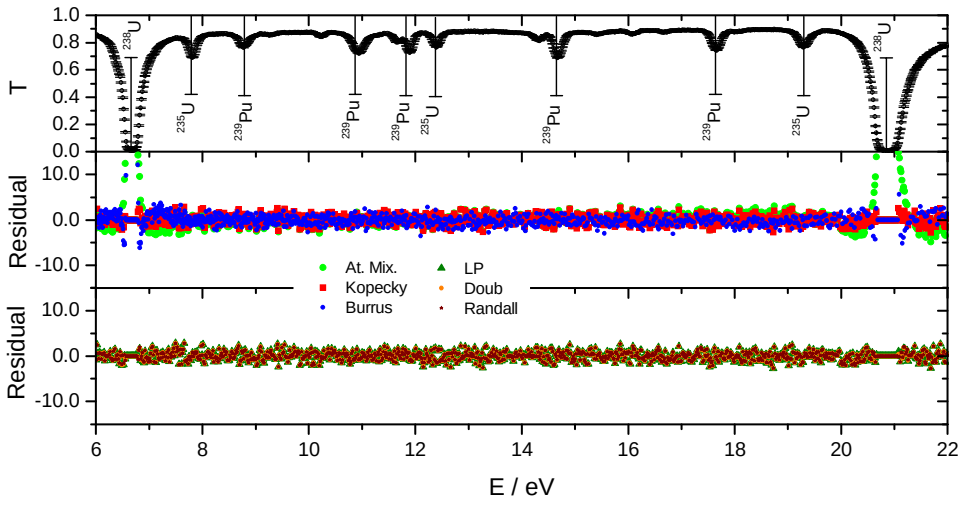


**Figure 5:** Created transmission spectrum and residual distribution in benchmark case 3.

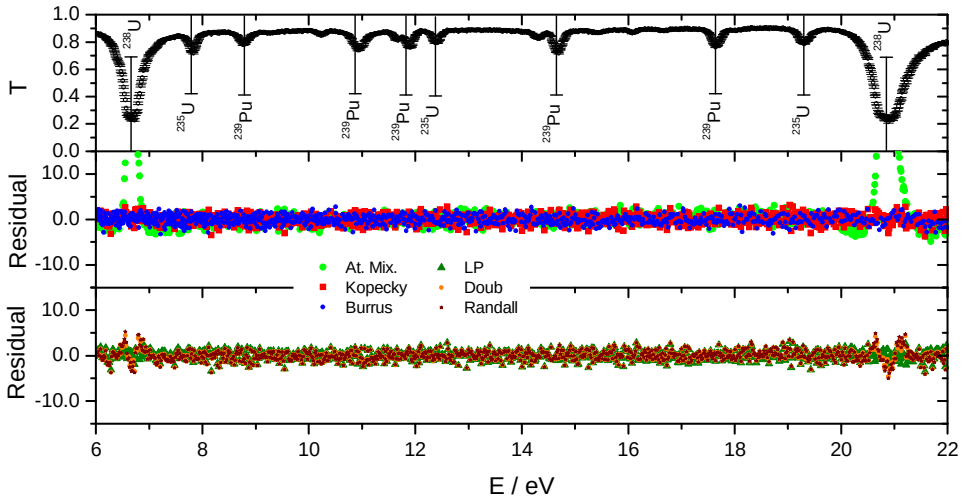


**Figure 6:** Created transmission spectrum and residual distribution in benchmark case 4.

## 4. Experimental Measurement Campaign at GELINA

Even though several analytical models can be used to determine accurately the $^{235}$U, $^{238}$U and $^{239}$Pu content by $\chi^2$ minimization in some of the benchmark cases, the question whether the MC realization reflects the transmission through a real powder sample remains. Therefore, several transmission measurements with powder samples are being made at the GELINA facility [22] of EC-JRC-IRMM. While in previous measurements of Doub [13], the transmission of thermal neutrons ($E < 1.23\,eV$) through a mixture of boron-carbide and aluminium spheres was studied, the measurements at EC-JRC-IRMM are dedicated to investigate the transmission at higher energies in the resolved resonance range. Samples made out of Cu and W powder mixed with S powder are being used. Transmission spectra of powder and solid metal samples are compared in order to be independent of cross section uncertainties. Table 2 gives an overview of the planned measurements.

## 5. Conclusion

Four different stochastic benchmark calculations, simulating the neutron transport through a powder sample, have been made to investigate the sensitivity of a NRD measurement on the inhomogeneity of the sample. Different analytical transport models have been used to determine the areal densities of $^{235}$U, $^{238}$U and $^{239}$Pu. It was shown that a simple homogenization of particles and matrix can result in a significant bias depending on the particle size. Depending on the benchmark case, several analytical models can be used to correctly determine the areal densities, i.e. to take into account the inhomogeneity of the sample. Transmission measurements through powder samples are currently being made at the GELINA facility of EC-JRC-IRMM in order to complement the numerical benchmark calculations. Based on these results an analytical method will be implemented in the REFIT code. The developed model and code will be used to study the inhomogeneity effect quantitatively for various conditions of sample.

| Sample | Eq. Thickness mm | Solid | Powder | Nominal Grain Size µm | Packing Fraction |
|---|---|---|---|---|---|
| Cu | 0.250 | x | | - | - |
| | | | x | 500 | 31% (pure Cu) |
| | | | x | 500 | 5.1% (Cu-S mix.) |
| | | | x | 150 | 5.1% (Cu-S mix.) |
| | 0.125 | x | | - | - |
| | | | x | 500 | 5.0% (Cu-S mix.) |
| | | | x | 150 | 5.1% (Cu-S mix.) |
| W | 0.150 | x | | - | - |
| | | | x | 50-250 | 14.5% (W-S mix.) |
| | 0.509 | | x | 50-250 | 17.0% (W-S mix.) |

**Table 2:** Planned powder transmission measurements at the GELINA facility of EC-JRC-IRMM.

## 6. Acknowledgements

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# Application of the PDET detector to BWR fuel assemblies: gross defect testing using the spatial distribution of neutron and photon flux

**Riccardo Rossa[1], Paolo Peerani[2], Young Ham[3], Shivakumar Sitaraman[3]**

[1] Studiecentrum voor Kernenergie–Centre d'etude de l'energie nucleaire (SCK-CEN); Université libre de Bruxelles (ULB). Formerly with JRC-ITU Ispra.
[2] Joint Research Centre – Institute for Transuranium Elements (JRC-ITU Ispra).
[3] Lawrence Livermore National Laboratory (LLNL).

Email: rrossa@sckcen.be

***Abstract:***

*Over 80 per cent of the material placed under safeguards today is in the form of spent fuel and one of the main ways to verify it is by Non-Destructive Assays. The main goal for the safeguards inspections is to verify that some or all the material has not been diverted to other purposes by detecting the eventual gross or partial defect.*
*The European Commission with the JRC-ITU located in Ispra in collaboration with Lawrence Livermore National Laboratory is studying the application of a detector to BWR fuel assemblies immersed in a spent fuel pool. The Partial Defect Tester (PDET) employs a set of neutron and photon detectors to be inserted in the fuel assembly to quantify the spatial distribution of the two fluxes within a PWR fuel assembly.*

*The insertion of multiple detectors inside a PWR assembly does not have a major technical problem and has been investigated by the Lawrence Livermore National Laboratory. The situation changes when the same concept is applied to a BWR assembly. A generic BWR assembly contains less fuel pins compared to a PWR assembly, but inside a traditional BWR assembly there are fewer water holes since the control rods have a cruciform shape and are inserted between neighbouring fuel assemblies.*

*The purpose of this study was to assess the feasibility of gross defect verification by evaluating the spatial distribution of both neutron and photon flux inside the BWR spent fuel assemblies. This study also gave an indication whether the irradiation history of the fuel assembly (burnup and cooling time) plays a role in the detection of the gross diversion.*
*The results showed that the gross defect is detectable by looking at the normalized ratio between the neutron and gamma signal (N/P ratio) as well as the normalized neutron and gamma signals. In particular, the change in shape of the normalized gamma signal appears to be a good indicator of a gross defect independent of operator declarations, regardless of whether the storage rack has assemblies with varying or non-varying burnups and cooling times.*

**Keywords:** PDET, spent fuel, gross defect, BWR, NDA

## 1. Introduction

Over 80 per cent of the material placed under safeguards today is in the form of spent fuel assemblies and one of the main ways to verify it is by Non-Destructive Assays (NDA) [1]. The technical goal for the safeguards inspections is to verify that all or some of the material has not been diverted to other purposes by detecting the eventual gross and partial defect from the fuel assemblies. Gross defect indicates the replacement of a complete fuel assembly with a dummy made of other material (e.g. natural uranium or stainless steel), while partial defect concerns the removal of fuel pins inside one assembly.

Most of the techniques currently used for safeguards inspection can reach a satisfactory level of accuracy only if the burnup and cooling time are precisely known, requiring that these data are provided by the operator [2], [3]. This is why several research centres and International agencies are putting significant effort to develop measurement methods that are able to detect missing material without relying on information provided by the operator.
The European Commission with the JRC-ITU located in Ispra is studying the application of a novel methodology based on the PDET tool, developed by the Lawrence Livermore National Laboratories (LLNL) for use with PWR assemblies, to BWR fuel assemblies stored in a spent fuel pond. PDET employs a set of tiny neutron and photon detectors to measure the respective signals at various spatial locations within a fuel assembly.
While absolute values of the signals will depend on burnup and cooling time, normalized ratios of these signals, individually or in combination, at the various measurement locations remain relatively invariant. The unique signature formed by this set of normalized signals is visibly distorted if fuel pins are missing from the assembly thus indicating material diversion. An experimental device has been designed, modelled and tested by LLNL with PWR assemblies [4],[5],[6] considering that this type of assembly allows the insertion of multiple detectors in the guide tubes already present in the geometry. The guide tubes are used for the insertion of the control rods during normal in-core operations and, once the assembly is stored in the spent fuel pool, they facilitate the insertion of small detectors (such as fission chambers or ion chambers).
The insertion of multiple detectors inside a BWR assembly is not as straightforward as the PWR case. A generic BWR assembly contains fewer fuel pins compared to a PWR assembly (e.g. 8x8 lattice instead of 17x17), with a larger diameter and pitch, resulting in similar external dimension (a square of about 16 cm versus the 22 cm of a PWR). However, inside the BWR assembly there are no guide tubes, because the control rods generally have a cruciform shape with blades that are inserted between neighbouring fuel elements. In some fuel geometries there are water holes used to have better neutron moderation and flattening of the intra-assembly flux during normal operations. However, these water holes are not accessible since they are closed at the top. In the case of a BWR fuel assembly, the detectors can typically be inserted only to the top of the first spacer

## 2. Model of the BWR fuel assembly

### 2.1. BWR fuel geometry

The reference model for this study will be the 8x8 ASEA BWR fuel [7] (Figure 1). This type of assembly has one central water hole. Considering the suggestions from LLNL a total of 38 measurement positions have been identified, both inside and around the fuel assembly [8]. These measurement locations are shown in Figure 1 and are depicted in orange, yellow, and white.

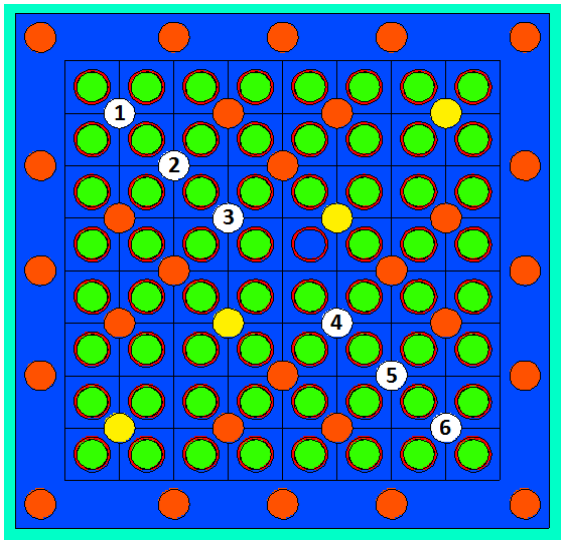


**Figure 1:** MCNP model of the BWR fuel assembly. Fuel pins are depicted in green, while measurement positions are reported in orange, yellow, and white.

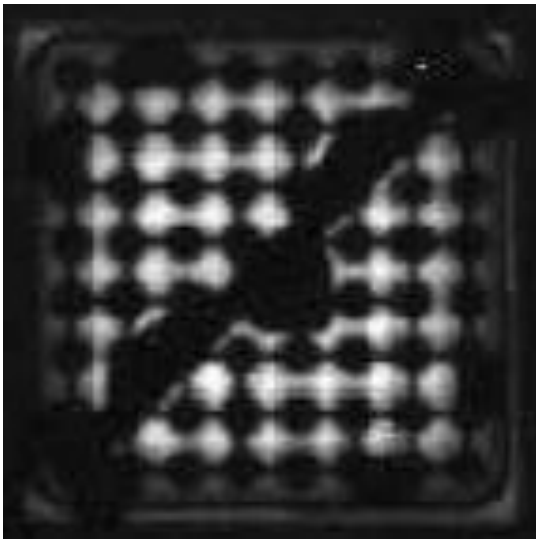

**Figure 2:** DCVD image of an ASEA 8x8 fuel assembly [9]. The black spots are the fuel pins, whereas the light part of the image is due to the Cherenkov light emitted in the surrounding water.

However, not all these positions can be used for the insertion of the PDET because the access is impeded by the handle at the top of the fuel. This item impedes access to 5 measurement points along one main diagonal (yellow points in Figure 1). This situation is confirmed by the image of the DCVD of an ASEA 8x8 BWR fuel assembly (Figure 2).
The main measurement points will be the six positions on the other main diagonal (points depicted in white in Figure 1) while the other channels (orange channels) will be used only if conclusive information cannot be obtained from the first measurements. No model for the detectors has been implemented in the geometry, assuming a constant efficiency factor. In order to introduce the detectors in the MCNP model [10], specific information regarding the dimensions of the instruments will be needed.

## 2.2. Spent fuel pond geometry

The storage rack for the study has been based on detailed information of the Swedish interim storage facility CLAB [11],[12]. This installation has 10 ponds containing storage baskets with spent fuel assemblies. There are three common types of baskets which are filled by both BWR and PWR assemblies. All the types of assemblies stored in CLAB can be inserted in all three types of baskets.

The geometry considered in this study is a basket that contains up to 16 BWR assemblies (Figure 3 represents the MCNP model). According to this configuration there is a water gap between the fuel and the basket, and the walls of the storage basket do not contain boron. By looking at Figure 1 and Figure 3 there is enough space in the basket to insert neutron and gamma detectors in the water gap outside the fuel assembly. Other baskets used in CLAB have borated steel in the internal walls, but all the external surfaces of the containers do not contain boron.

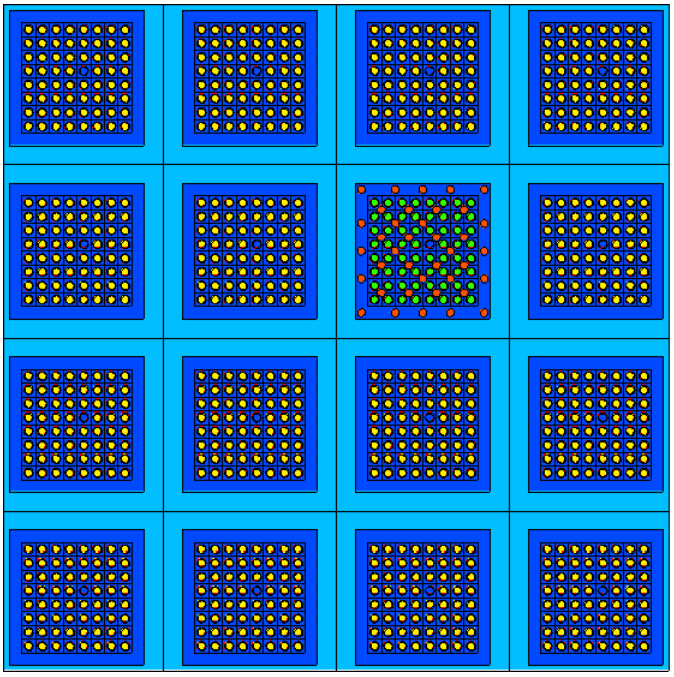

**Figure 3:** model of the storage rack.
The fuel assembly under investigation is depicted in green and has the measurement positions coloured in orange.

## 2.3. Spent fuel compositions

The spent fuel compositions have been calculated with ORIGEN-ARP [13] under the conditions:

- Initial Enrichment (IE): 3%
- Burnup (BU): 14 – 28 – 42 GWd/tU
- Cooling Time (CT): 1 – 30 years

Each irradiation cycle lasted 350 days and two successive irradiation cycles have been separated by 30 days of cooling time. Adjusting the power level to 40 MW/tU, each irradiation cycle resulted in a burnup of 14 GWd/tU.
Considering that the cooling time had a very small influence on the detectors' response, only two values for that parameter have been considered.
In order to easily identify the characteristics of the fuel assemblies used for the simulations, they are identified by four numbers: the first two represents the burnup in GWd/tU while the other two are the years of cooling time after discharge. This convention can be seen in the first column of Table 1.
The material composition of spent fuel used in the Monte Carlo simulations included the 37 principal absorbers (both fission products and minor actinides) [14].

## 2.4. Neutron and photon source terms

Neutrons are generated within all pins containing spent fuel in all the fuel assemblies. The source intensity is proportional to the total neutron emission of the assembly; value that is taken from the ORIGEN-ARP simulations (Table 1). All pins within a fuel assembly have the same source intensity.
The spectrum used for the neutron source is a Watt spectrum with parameters of Cm-244 [9], which accounts for over 90% of the total neutron source in the spent fuel.
As in the case of neutron generation, photons are generated taking into account the total gamma emission of the fuel pin. All pins within a fuel assembly have the same source intensity. The photon source has been modelled as a line at 662 keV, the main emission of Cs-137, which accounts for over 80% of the total photon source in the spent fuel after few years of cooling.

| **ID assembly** | **Neutron source** (n/s/tU) | **Photon source** (pn/s/tU) |
|---|---|---|
| 1401 | 6.741E+06 | 1.290E+16 |
| 1430 | 2.851E+06 | 8.289E+14 |
| 2801 | 2.071E+08 | 2.258E+16 |
| 2830 | 6.111E+07 | 1.692E+15 |
| 4201 | 7.843E+08 | 3.160E+16 |
| 4230 | 2.432E+08 | 2.470E+15 |

**Table 1:** summary of the neutron and photon source terms of the fuel assemblies used in the simulations.

## 2.5. Results from the reference case

The first set of simulations concerned the 4x4 lattice filled with fuel assemblies having the same combination of initial enrichment, burnup, and cooling time. The next tables (Table 2.a and Table 2.b) summarize the results from the simulations. Table 2.a reports the neutron and gamma tallies calculated in the reference cases. Since multiple "virtual" detectors were placed within the fuel assembly, each tally has been normalized by the maximum value found in the simulation. It is evident that both normalized fluxes are not affected by the composition of the fuel assembly (Figures 4 and 5).
Table 2.b reports the N/P ratios and the normalized ratios calculated for the same simulations. The burnup and the cooling time affect the ratio in a significant way since the N/P ratio changes of two orders of magnitude from 0.29 to 33.41. On the other hand the normalized N/P ratio does not vary substantially for the different cases.

| **REF** | **1401** | | **1430** | | **2810** | | **4201** | | **4230** | |
|---|---|---|---|---|---|---|---|---|---|---|
| Pos. | N/P | Norm. N/P | N/P | Norm. N/P | N/P | Norm. N/P | N/P | Norm. N/P | N/P | Norm. N/P |
| 1 | 0.33 | 1.00 | 2.07 | 1.00 | 19.73 | 1.00 | 9.31 | 1.00 | 33.41 | 1.00 |
| 2 | 0.30 | 0.90 | 1.86 | 0.90 | 17.43 | 0.88 | 8.33 | 0.89 | 29.81 | 0.89 |
| 3 | 0.29 | 0.89 | 1.83 | 0.88 | 17.19 | 0.87 | 8.11 | 0.87 | 29.13 | 0.87 |
| 4 | 0.29 | 0.89 | 1.85 | 0.90 | 17.39 | 0.88 | 8.29 | 0.89 | 29.91 | 0.90 |
| 5 | 0.29 | 0.89 | 1.85 | 0.89 | 17.33 | 0.88 | 8.29 | 0.89 | 29.79 | 0.89 |
| 6 | 0.32 | 0.99 | 2.07 | 1.00 | 19.44 | 0.99 | 9.19 | 0.99 | 32.87 | 0.98 |

**Table 2.b:** summary of the results in the reference case.

| | 1401 - REF | | | | 1430 - REF | | | | 2810 - REF | | | | 4201 - REF | | | | 4230 - REF | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Pos | Neutron tally | Norm N | Photon tally | Norm P | Neutron tally | Norm N | Photon tally | Norm P | Neutron tally | Norm N | Photon tally | Norm P | Neutron tally | Norm N | Photon tally | Norm P | Neutron tally | Norm N | Photon tally | Norm P |
| 1 | 6.915E-06 | 0.86 | 2.108E-05 | 0.76 | 2.807E-06 | 0.86 | 1.355E-06 | 0.76 | 9.903E-05 | 0.87 | 5.018E-06 | 0.76 | 4.828E-04 | 0.86 | 5.183E-05 | 0.76 | 1.353E-04 | 0.86 | 4.051E-06 | 0.76 |
| 2 | 7.623E-06 | 0.95 | 2.583E-05 | 0.93 | 3.093E-06 | 0.94 | 1.660E-06 | 0.93 | 1.070E-04 | 0.94 | 6.142E-06 | 0.93 | 5.282E-04 | 0.94 | 6.344E-05 | 0.93 | 1.478E-04 | 0.94 | 4.959E-06 | 0.93 |
| 3 | 8.053E-06 | 1.00 | 2.774E-05 | 1.00 | 3.263E-06 | 1.00 | 1.782E-06 | 1.00 | 1.134E-04 | 1.00 | 6.600E-06 | 1.00 | 5.527E-04 | 0.99 | 6.819E-05 | 1.00 | 1.553E-04 | 0.98 | 5.330E-06 | 1.00 |
| 4 | 8.045E-06 | 1.00 | 2.747E-05 | 0.99 | 3.274E-06 | 1.00 | 1.765E-06 | 0.99 | 1.137E-04 | 1.00 | 6.538E-06 | 0.99 | 5.594E-04 | 1.00 | 6.752E-05 | 0.99 | 1.579E-04 | 1.00 | 5.278E-06 | 0.99 |
| 5 | 7.578E-06 | 0.94 | 2.596E-05 | 0.94 | 3.086E-06 | 0.94 | 1.668E-06 | 0.94 | 1.071E-04 | 0.94 | 6.177E-06 | 0.94 | 5.289E-04 | 0.95 | 6.381E-05 | 0.94 | 1.486E-04 | 0.94 | 4.988E-06 | 0.94 |
| 6 | 6.869E-06 | 0.85 | 2.119E-05 | 0.76 | 2.818E-06 | 0.86 | 1.362E-06 | 0.76 | 9.803E-05 | 0.86 | 5.042E-06 | 0.76 | 4.787E-04 | 0.86 | 5.208E-05 | 0.76 | 1.338E-04 | 0.85 | 4.071E-06 | 0.76 |

**Table 2.a:** summary of the results in the reference case.

| | 1401 - GSS | | | | 1430 - GSS | | | | 2810 - GSS | | | | 4201 - GSS | | | | 4230 - GSS | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Pos | Neutron tally | Norm N | Photon tally | Norm P | Neutron tally | Norm N | Photon tally | Norm P | Neutron tally | Norm N | Photon tally | Norm P | Neutron tally | Norm N | Photon tally | Norm P | Neutron tally | Norm N | Photon tally | Norm P |
| 1 | 2.071E-06 | 1.00 | 1.644E-06 | 0.99 | 8.540E-07 | 1.00 | 1.057E-07 | 0.99 | 3.122E-05 | 1.00 | 3.914E-07 | 1.00 | 1.573E-04 | 1.00 | 4.041E-06 | 1.00 | 4.469E-05 | 1.00 | 3.159E-07 | 1.00 |
| 2 | 1.716E-06 | 0.83 | 1.393E-06 | 0.84 | 7.029E-07 | 0.82 | 8.951E-08 | 0.84 | 2.585E-05 | 0.83 | 3.314E-07 | 0.84 | 1.285E-04 | 0.82 | 3.419E-06 | 0.84 | 3.673E-05 | 0.82 | 2.673E-07 | 0.84 |
| 3 | 1.518E-06 | 0.73 | 1.195E-06 | 0.72 | 6.208E-07 | 0.73 | 7.677E-08 | 0.72 | 2.307E-05 | 0.74 | 2.840E-07 | 0.72 | 1.153E-04 | 0.73 | 2.932E-06 | 0.72 | 3.279E-05 | 0.73 | 2.292E-07 | 0.72 |
| 4 | 1.503E-06 | 0.73 | 1.179E-06 | 0.71 | 6.188E-07 | 0.72 | 7.578E-08 | 0.71 | 2.262E-05 | 0.72 | 2.803E-07 | 0.71 | 1.114E-04 | 0.71 | 2.896E-06 | 0.71 | 3.205E-05 | 0.72 | 2.263E-07 | 0.71 |
| 5 | 1.692E-06 | 0.82 | 1.375E-06 | 0.83 | 6.957E-07 | 0.81 | 8.834E-08 | 0.83 | 2.557E-05 | 0.82 | 3.271E-07 | 0.83 | 1.279E-04 | 0.81 | 3.377E-06 | 0.83 | 3.681E-05 | 0.82 | 2.639E-07 | 0.83 |
| 6 | 2.052E-06 | 0.99 | 1.653E-06 | 1.00 | 8.359E-07 | 0.98 | 1.062E-07 | 1.00 | 3.099E-05 | 0.99 | 3.924E-07 | 1.00 | 1.557E-04 | 0.99 | 4.056E-06 | 1.00 | 4.440E-05 | 0.99 | 3.170E-07 | 1.00 |

**Table 3.a:** summary of the results of the diversion case with a stainless steel dummy.

| | 1401 - GNU | | | | 1430 - GNU | | | | 2810 - GNU | | | | 4201 - GNU | | | | 4230 - GNU | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Pos | Neutron tally | Norm N | Photon tally | Norm P | Neutron tally | Norm N | Photon tally | Norm P | Neutron tally | Norm N | Photon tally | Norm P | Neutron tally | Norm N | Photon tally | Norm P | Neutron tally | Norm N | Photon tally | Norm P |
| 21 | 3.401E-06 | 1.00 | 7.312E-07 | 1.00 | 1.391E-06 | 1.00 | 4.698E-08 | 1.00 | 5.109E-05 | 1.00 | 1.739E-07 | 1.00 | 2.515E-04 | 1.00 | 1.795E-06 | 1.00 | 7.138E-05 | 1.00 | 1.403E-07 | 1.00 |
| 42 | 3.332E-06 | 0.98 | 3.665E-07 | 0.50 | 1.371E-06 | 0.99 | 2.355E-08 | 0.50 | 4.994E-05 | 0.98 | 8.731E-08 | 0.50 | 2.489E-04 | 0.99 | 9.000E-07 | 0.50 | 7.088E-05 | 0.99 | 7.035E-08 | 0.50 |
| 26 | 3.257E-06 | 0.96 | 2.257E-07 | 0.31 | 1.337E-06 | 0.96 | 1.450E-08 | 0.31 | 4.872E-05 | 0.95 | 5.354E-08 | 0.31 | 2.415E-04 | 0.96 | 5.541E-07 | 0.31 | 6.822E-05 | 0.95 | 4.331E-08 | 0.31 |
| 31 | 3.286E-06 | 0.97 | 2.304E-07 | 0.32 | 1.348E-06 | 0.97 | 1.480E-08 | 0.32 | 4.933E-05 | 0.97 | 5.479E-08 | 0.32 | 2.424E-04 | 0.96 | 5.661E-07 | 0.32 | 6.926E-05 | 0.97 | 4.425E-08 | 0.32 |
| 45 | 3.322E-06 | 0.98 | 3.645E-07 | 0.50 | 1.354E-06 | 0.97 | 2.342E-08 | 0.50 | 4.951E-05 | 0.97 | 8.695E-08 | 0.50 | 2.458E-04 | 0.98 | 8.976E-07 | 0.50 | 6.973E-05 | 0.98 | 7.016E-08 | 0.50 |
| 36 | 3.389E-06 | 1.00 | 7.133E-07 | 0.98 | 1.389E-06 | 1.00 | 4.583E-08 | 0.98 | 5.093E-05 | 1.00 | 1.696E-07 | 0.98 | 2.512E-04 | 1.00 | 1.752E-06 | 0.98 | 7.145E-05 | 1.00 | 1.370E-07 | 0.98 |

**Table 4.a:** summary of the results of the diversion case with a natural uranium dummy.

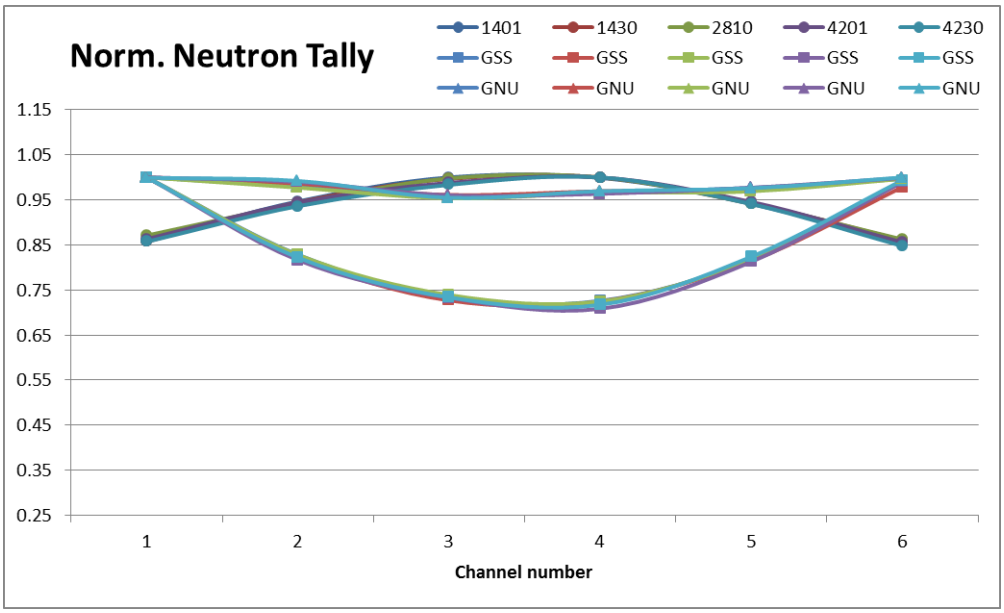


**Figure 4:** comparison of the normalized neutron tally along the channels in the main diagonal.

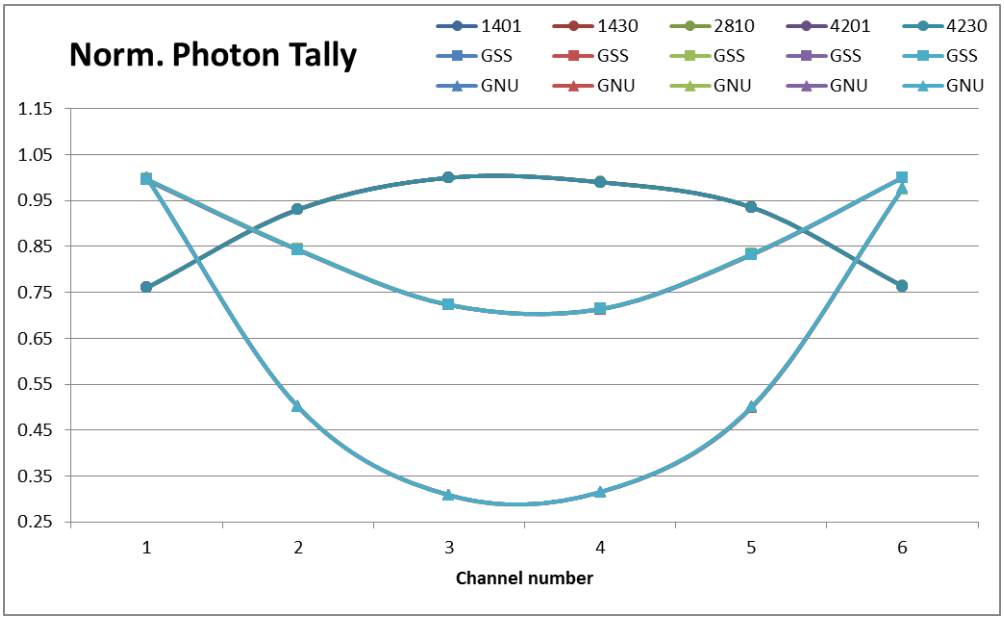


**Figure 5:** comparison of the normalized photon tally along the channels in the main diagonal.

## 2.6. Results from the diversion scenarios

Starting from the reference case the gross defect scenarios have been evaluated using the same combinations of burnup and cooling time. This time one of the central assemblies in the lattice has been replaced by a dummy assembly of stainless steel or natural uranium.

Following the same procedure as before, both neutron and photon fluxes have been computed using the flux tallies in the six diagonal channels. Results are shown in Tables 3 (case with the stainless steel dummy, GSS), and Tables 4 (natural uranium dummy, GNU).

Figures 4 and 5 show the change in the concavity of the neutron and gamma flux profiles in the diversion scenarios. Especially in the case of gamma signals, the gross defect can be easily detected by comparing the profile of the normalized flux along the measurement positions.

This set of simulation also confirmed the increase on the N/P ratio in the case of replacement with natural uranium dummy. This is due to the strong photon shielding due to the natural uranium, while the neutrons are less attenuated by a low-Z material such as steel.

By looking at the absolute values of the fluxes (Tables 2.a, 3.a, 4.a), an inspector can see the huge difference between the reference cases and the diversion scenarios. Especially the photon flux decreases of about one order of magnitude when the central assembly is removed. This comparison however relies on the operator's data of burnup and cooling time.

| GSS | 1401 | | 1430 | | 2810 | | 4201 | | 4230 | |
|---|---|---|---|---|---|---|---|---|---|---|
| Pos. | N/P | Norm. N/P | N/P | Norm. N/P | N/P | Norm. N/P | N/P | Norm. N/P | N/P | Norm. N/P |
| 1 | 1.26 | 0.99 | 8.08 | 0.99 | 79.78 | 0.98 | 38.93 | 0.99 | 141.46 | 0.99 |
| 2 | 1.23 | 0.97 | 7.85 | 0.96 | 78.00 | 0.96 | 37.57 | 0.96 | 137.42 | 0.96 |
| 3 | 1.27 | 1.00 | 8.09 | 0.99 | 81.25 | 1.00 | 39.33 | 1.00 | 143.07 | 1.00 |
| 4 | 1.27 | 1.00 | 8.17 | 1.00 | 80.72 | 0.99 | 38.48 | 0.98 | 141.59 | 0.99 |
| 5 | 1.23 | 0.97 | 7.88 | 0.96 | 78.18 | 0.96 | 37.87 | 0.96 | 139.47 | 0.97 |
| 6 | 1.24 | 0.97 | 7.87 | 0.96 | 78.97 | 0.97 | 38.39 | 0.98 | 140.05 | 0.98 |

**Table 3.b:** summary of the results of the diversion case with a stainless steel dummy.

| GNU | 1401 | | 1430 | | 2810 | | 4201 | | 4230 | |
|---|---|---|---|---|---|---|---|---|---|---|
| Pos. | N/P | Norm. N/P | N/P | Norm. N/P | N/P | Norm. N/P | N/P | Norm. N/P | N/P | Norm. N/P |
| 1 | 4.65 | 0.32 | 29.61 | 0.32 | 293.84 | 0.32 | 140.11 | 0.32 | 508.67 | 0.32 |
| 2 | 9.09 | 0.63 | 58.24 | 0.63 | 571.97 | 0.63 | 276.62 | 0.63 | 1007.66 | 0.64 |
| 3 | 14.43 | 1.00 | 92.18 | 1.00 | 910.06 | 1.00 | 435.88 | 1.00 | 1575.05 | 1.00 |
| 4 | 14.26 | 0.99 | 91.10 | 0.99 | 900.41 | 0.99 | 428.08 | 0.98 | 1565.05 | 0.99 |
| 5 | 9.11 | 0.63 | 57.83 | 0.63 | 569.46 | 0.63 | 273.81 | 0.63 | 993.79 | 0.63 |
| 6 | 4.75 | 0.33 | 30.30 | 0.33 | 300.38 | 0.33 | 143.38 | 0.33 | 521.62 | 0.33 |

**Table 4.b:** summary of the results of the diversion case with a natural uranium dummy.

Figure 6 shows the comparison of the normalized N/P ratio signal coming from the six positions along the main diagonal. As suggested from work performed at LLNL, the generic reference case has the shape of an upward parabola, while the replacement with a stainless steel dummy (GSS) gives constant values of the ratio and replacement with natural uranium (GNU) gives a downward parabola. Therefore this study suggests that for uniform pattern in the storage rack the gross diversion can be detected without the support of operator's data. The next section will investigate the case of a generic patter using assemblies with different burnup and cooling time.

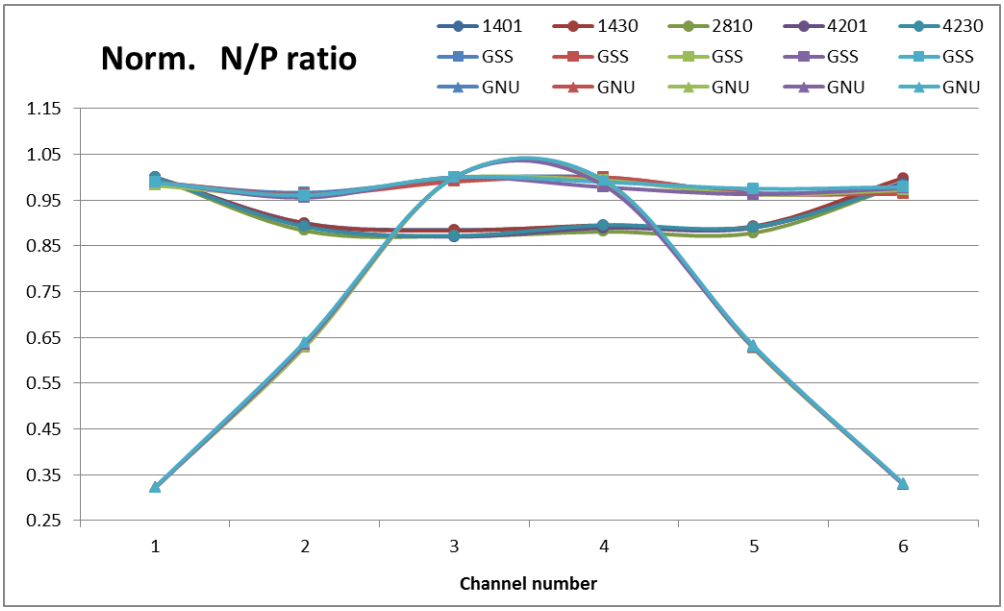


**Figure 6:** comparison of the normalized N/P ratio along the channels in the main diagonal.

## 2.7. Simulation of a generic rack pattern

The last part of the study simulated a generic rack pattern, using fuel assemblies with a combination of burnup and cooling time. This is done to examine the response of the detectors in the most general condition. The configuration of the storage basket is reported in Figure 7. Following the convention explained in section 2.3, each assembly is identified by four numbers: the first two are the burnup expressed in GWd/tU while the other two are the years of cooling time. The position with a yellow background is the assembly tested for the diversion scenario.

| | | | |
|---|---|---|---|
| 2810 | 1401 | 1430 | 1430 |
| 2810 | 2810 | 2810 | 4201 |
| 2810 | 1401 | 4201 | 1401 |
| 2810 | 2810 | 1401 | 1430 |

**Figure 7:** composition of the storage rack. The detectors are inserted in the position with the yellow background.

The simulations considered again the reference case (REF) and the diversion of the tested assembly with a dummy made of stainless steel (GSS) or natural uranium (GNU). Tables 5-7 show the numerical results of the Monte Carlo simulations, while Figures 8-10 are the plots of the normalized tallies and N/P ratio for the same cases.
As in the case of a uniform storage basket both the neutron and photon tally are clearly lower in case of a gross diversion. The profile of the photon tally and also the normalized N/P ratio is significantly distorted from the reference case, providing a check on the operator's data and an indication for the detection of a gross defect.

| | Pos. | Neutron tally | Norm N | Photon tally | Norm P | N/P | Norm N/P |
|---|---|---|---|---|---|---|---|
| **Generic - REF** | 1 | 9.278E-05 | 0.61 | 1.812E-06 | 0.72 | 51.21 | 0.80 |
| | 2 | 1.072E-04 | 0.71 | 2.208E-06 | 0.88 | 48.57 | 0.76 |
| | 3 | 1.228E-04 | 0.81 | 2.394E-06 | 0.95 | 51.31 | 0.80 |
| | 4 | 1.442E-04 | 0.95 | 2.509E-06 | 1.00 | 57.46 | 0.90 |
| | 5 | 1.487E-04 | 0.98 | 2.498E-06 | 1.00 | 59.56 | 0.93 |
| | 6 | 1.521E-04 | 1.00 | 2.372E-06 | 0.95 | 64.10 | 1.00 |

**Table 5:** results from the testing of the generic pattern of the storage basket – reference case.

| | Pos. | Neutron tally | Norm N | Photon tally | Norm P | N/P | Norm N/P |
|---|---|---|---|---|---|---|---|
| **Generic - GSS** | 1 | 2.409E-05 | 0.30 | 2.252E-07 | 0.19 | 106.96 | 1.00 |
| | 2 | 2.551E-05 | 0.32 | 2.699E-07 | 0.22 | 94.51 | 0.88 |
| | 3 | 2.948E-05 | 0.37 | 3.636E-07 | 0.30 | 81.07 | 0.76 |
| | 4 | 4.672E-05 | 0.59 | 6.754E-07 | 0.56 | 69.18 | 0.65 |
| | 5 | 6.146E-05 | 0.77 | 9.295E-07 | 0.77 | 66.11 | 0.62 |
| | 6 | 7.945E-05 | 1.00 | 1.203E-06 | 1.00 | 66.02 | 0.62 |

**Table 6:** results from the testing of the generic pattern of the storage basket – replacement with stainless steel dummy.

| | Pos. | Neutron tally | Norm N | Photon tally | Norm P | N/P | Norm N/P |
|---|---|---|---|---|---|---|---|
| **Generic - GNU** | 1 | 5.154E-05 | 0.44 | 6.353E-08 | 0.11 | 811.28 | 0.65 |
| | 2 | 6.041E-05 | 0.51 | 4.869E-08 | 0.09 | 1240.55 | 1.00 |
| | 3 | 7.098E-05 | 0.60 | 6.201E-08 | 0.11 | 1144.75 | 0.92 |
| | 4 | 9.497E-05 | 0.81 | 1.482E-07 | 0.26 | 640.66 | 0.52 |
| | 5 | 1.069E-04 | 0.91 | 2.748E-07 | 0.48 | 388.99 | 0.31 |
| | 6 | 1.178E-04 | 1.00 | 5.671E-07 | 1.00 | 207.79 | 0.17 |

**Table 7:** results from the testing of the generic pattern of the storage basket – replacement with natural uranium dummy.

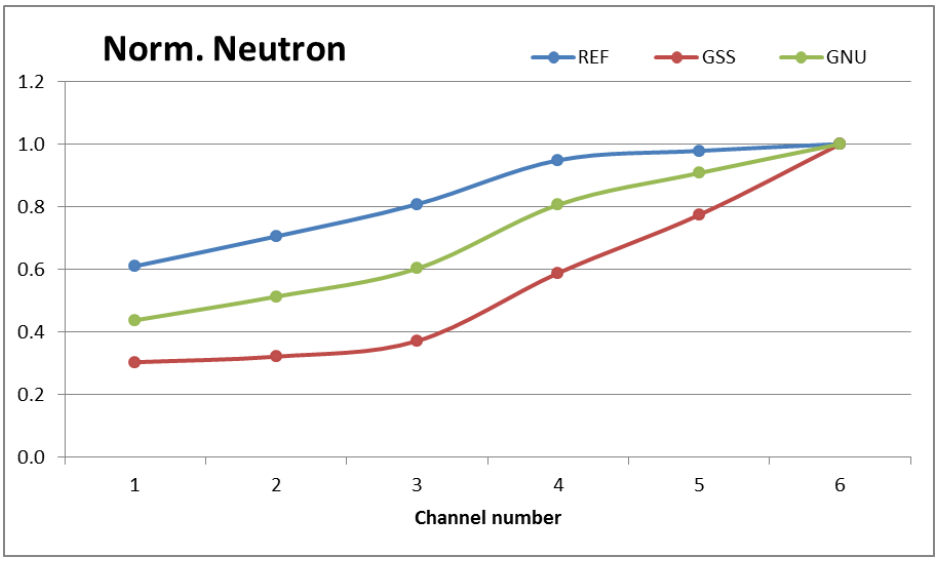


**Figure 8:** comparison of the normalized neutron tally for the generic pattern of the storage basket.

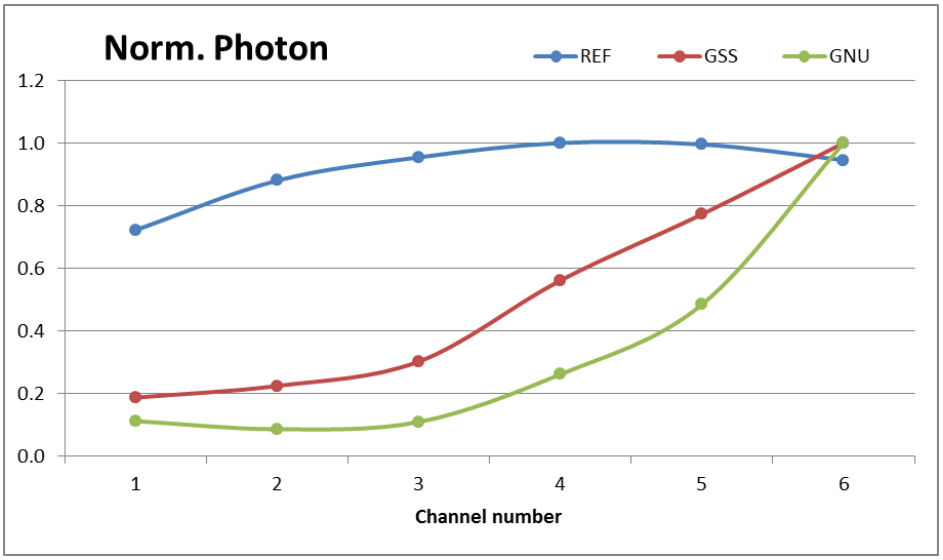


**Figure 9:** comparison of the normalized photon tally for the generic pattern of the storage basket.

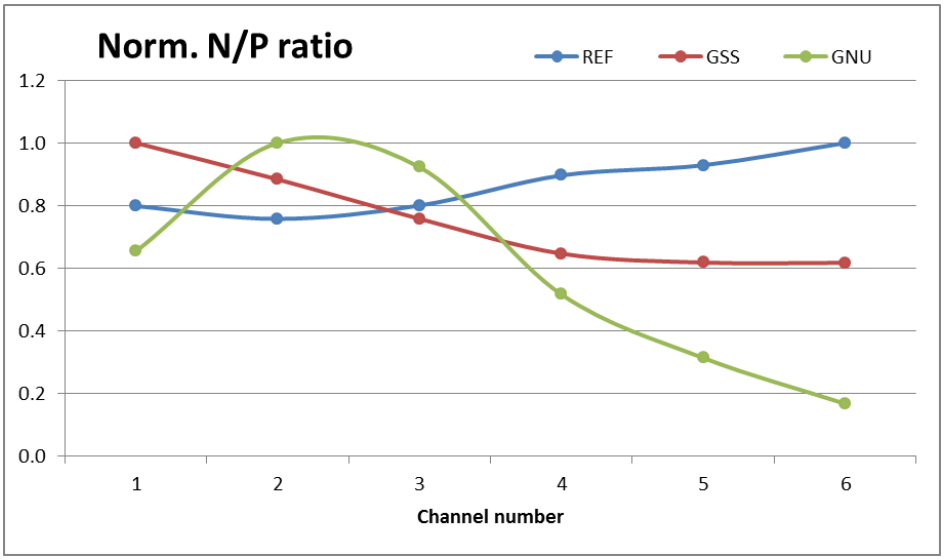


**Figure 10:** comparison of the normalized N/P ratio for the generic pattern of the storage basket.

## Conclusions and future work

This report provides a study on gross defect testing using a technique similar to the one used by PDET for PWR spent fuel applied to BWR fuel assemblies. The paper considered the use of several intra-assembly measurement positions to reconstruct the neutron and photon flux profile in the assembly. Considering the information coming from CLAB and LLNL, the neutron and gamma detectors can be inserted in a total of 38 positions both inside and outside the fuel assembly, but only 6 of them were investigated in detail. These positions are along the main diagonal of the assembly.
The first set of simulations considered a uniform storage rack with fuel assemblies with the same burnup and cooling time. In these cases it is possible to detect a gross diversion by the distortion of the gamma signal and of the normalized N/P ratio along the diagonal. The profile of the normalized gamma tallies along the main diagonal of the assembly is of a downward parabola in the reference case, while the concavity is inverted in the case of gross defect. On the other hand the shape of the normalized N/P ratio is of an upward parabola, while in the diversion scenarios the concavity is inverted or the shape is constant for all positions.
The last set of simulations represented a more general pattern in the storage rack with assemblies of different burnup and cooling time. In this case also, the diversion can be detected by the different behaviour of the normalized gamma flux and normalized N/P ratio of the detectors. This is especially true for the normalized gamma profiles that maintain the same downward parabolic shape as in the uniform case for the reference case that gets distorted in the diverted case..
Considering the promising results that the technique has achieved both in the investigation with PWR and BWR assemblies, future work will address specific issues that arose in this feasibility study. In particular there will be an effort to quantify the possible sources of uncertainty in the calculation for the three normalized ratios and investigate other possible positions for the detectors. Moreover future simulations will integrate the detectors themselves in the fuel geometry to consider the impact that the instruments will have on the neutron and gamma profiles.

## Acknowledgements

This work has been performed in the frame of the Action Sheet 39 under the Euratom-DoE Agreement. The contribution from Lawrence Livermore National Laboratory was performed under the auspices of the U.S. Department of Energy under Contract DE-AC52-07NA27344. The authors wish to thank the NNSA Next Generation Safeguards Initiative Program for their generous support of this project.

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# Development of Neutron Resonance Densitometry at the GELINA TOF Facility

**P. Schillebeeckx[1], S. Abousahl[2], B. Becker[1], A. Borella[3], F. Emiliani[1], H. Harada[4], K. Kauwenberghs[1], F. Kitatani[4], M. Koizumi[4], S. Kopecky[1], A. Moens[1], M. Moxon[5], G. Sibbens[1] and H. Tsuchiya[4]**

[1]EC-JRC-IRMM
Unit D.4. Standards for Nuclear Safety, Security and Safeguards,
Retieseweg 111, B- 2440 Geel, Belgium
E-mail: peter.schillebeeckx@ec.europa.eu

[2]EC-JRC-HQ
Marsveldstraat 21, B- 1050 Brussels, Belgium

[3]SCK•CEN
Boeretang 200, B- 2400 Mol, Belgium

[4]Japan Atomic Energy Agency (JAEA)
Tokai-mura, Naka-gun, Ibaraki 319 – 1195, Japan

[5]Hyde Copse 3,
Marcham, United Kingdom

***Abstract:***

*Neutrons can be used as a tool to study properties of materials and objects. An evolving activity in this field concerns the existence of resonances in neutron induced reaction cross sections. These resonance structures are the basis of two analytical methods which have been developed at the EC – JRC – IRMM: Neutron Resonance Capture Analysis (NRCA) and Neutron Resonance Transmission Analysis (NRTA). They have been applied to determine the elemental composition of archaeological objects and to characterize nuclear reference materials.*

*A combination of NRTA and NRCA together with Prompt Gamma Neutron Analysis, referred to as Neutron Resonance Densitometry (NRD), is being studied as a non-destructive method to characterize particle-like debris of melted fuel that is formed in severe nuclear accidents such as the one which occurred at the Fukushima Daiichi nuclear power plants. This study is part of a collaboration between JAEA and EC – JRC – IRMM.*

*In this contribution the basic principles of NRTA and NRCA are explained based on the experience in the use of these methods at the time-of-flight facility GELINA of the EC – JRC – IRMM. Specific problems related to the analysis of samples resulting from melted fuel are discussed. The programme to study and solve these problems is described and results of a first measurement campaign at GELINA are given.*

**Keywords:** non-destructive assay; time-of-flight; resonance analysis; melted fuel; severe accidents; nuclear safeguards; transmission; capture; GELINA

## 1. Introduction

The probability that a neutron interacts with nuclei strongly depends on the energy of the neutron. This is shown in Fig.1, which compares the total cross section as a function of the kinetic energy of the interacting neutron for several nuclides. The cross sections reveal

the presence of resonance structures. The origin of these structures is well understood. The resonances are related to excited states of the compound nucleus which is formed by the neutron and the target nucleus. The resonance structured cross sections can be parameterized by resonance parameters based on the R-matrix nuclear reaction formalism [1]. Each resonance is characterized by a set of resonance parameters, in particular the resonance energy and the partial reaction widths. The partial widths (e.g. the neutron, capture and fission width) express the probability for a specific reaction to occur. The smooth part of the total cross section is due to scattering from the nuclear potential and its magnitude depends on the scattering radius. Since resonances are observed at energies which are specific for each nuclide they can be used as fingerprints to determine the elemental and even isotopic composition of materials and objects [2,3]. The resonance structures in the total and capture cross sections are the basis of Neutron Resonance Transmission (NRTA) and Neutron Resonance Capture Analysis (NRCA), respectively. Both NRCA and NRTA are non-destructive methods, which determine the bulk elemental composition, do not require any sample preparation and result in a negligible residual activation. In this contribution the basic principles of NRTA and NRCA are discussed, with a special emphasis on the use of NRTA as an absolute method to determine the amount of fissile and fertile materials for safeguards applications.

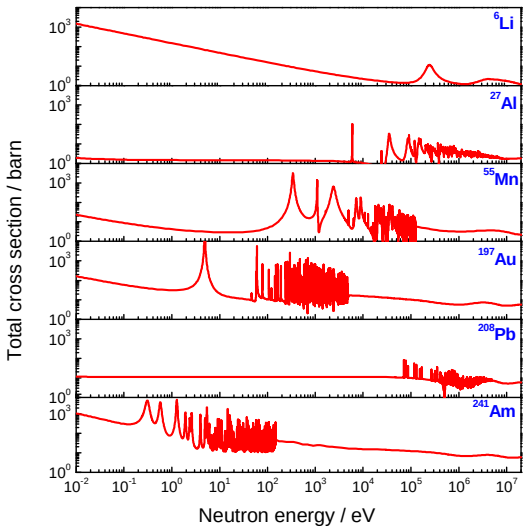


**Figure 1:**Total cross section as a function of neutron energy for neutron induced reactions in $^{6}$Li, $^{27}$Al, $^{55}$Mn, $^{197}$Au, $^{208}$Pb and $^{241}$Am.

## 2. Basic principles of NRTA and NRCA

Both NRTA and NRCA are based on the time-of-flight (TOF) technique, which is a standard technique for neutron resonance spectroscopy. They rely on the same principles and methods as those used for the determination of cross section data in the resonance region. These principles and methods, including the production of full covariance data, have been reviewed recently in Ref. [4]. NRTA is based on the analysis of characteristic dips in a transmission spectrum that can be obtained from a measurement of the attenuation of the neutron beam by a sample [2,3]. These dips are observed at TOF values corresponding to resonance energies. NRCA refers to the analysis of resonance structures in TOF spectra obtained from the detection of prompt γ-rays, which are emitted after a neutron capture reaction in the sample [2,3]. In Ref. [5] a comparison of NRCA and PGAA (Prompt Gamma-ray Activation Analysis), which is also based on the detection of prompt γ-rays, is reported.

### 2.1. Time-of-flight measurements

A precise knowledge of the energy of the neutron inducing a reaction in the sample is required to make use of the resonance structure in neutron induced reaction cross sections for material analysis. For a quantitative analysis covering a wide range of elements, time-of-flight measurements at an accelerator-based pulsed white neutron source are preferred. Pulsed neutron sources can be realized using electron- and proton-based accelerators. In electron-based accelerators, high-energy electrons generate Bremsstrahlung in a target and neutrons are produced via photonuclear reactions. High-energy proton accelerators produce neutrons via the spallation process in a target made out of high mass number material. The energy spectrum of neutrons produced by photonuclear reactions or the spallation process is not directly exploitable for low energy resonance spectroscopy. Therefore, a moderator containing e.g. hydrogen rich material is used to increase the amount of low-energy neutrons and to produce a broad neutron spectrum ranging from thermal energies up to the high energy region.

Experimentally, the time-of-flight $t_m$ is derived from the difference between a stop ($T_s$) and a start signal ($T_0$). The start signal is produced by the pulsed charged particle beam. The stop

signal in a transmission experiment (NRTA) is provided by the neutron detector. In a capture experiment (NRCA) the arrival time is obtained from the detection of the reaction products which are emitted in the neutron induced reaction. The time-of-flight $t$ that a neutron needs to travel a distance $L$ can be related to the velocity $v$ of the neutron at the moment it leaves the neutron producing target and enters the detector or sample:

$$v = \frac{L}{t} = \frac{L}{t_m - (t_t + t_d)} \quad (1)$$

where $t_t$ is the time the neutron spends in the neutron producing target and $t_d$ the time spend in the neutron detector or sample. Both $t_t$ and $t_d$ can depend on the kinetic energy of the neutron E given by:

$$E = m\,c^2(\gamma - 1) \quad (2)$$

where $c$ denotes the speed of light, $m$ the rest mass of the neutron and $\gamma$ the Lorentz factor.

The resolution $\Delta v$ which is a combination of the broadening due to the finite width of the timing channels and the spread due to the time the neutrons spend in the source and detector, can be described by broadening due to the time-of-flight ($\Delta t$) and distance ($\Delta L$):

$$\frac{\Delta v}{v} = \frac{1}{L}\sqrt{(v\Delta t)^2 + \Delta L^2} \quad (3).$$

The resolution improves with increasing distance. The distance $L$ can be determined by metric measurements with an uncertainty smaller than 1 mm. The contribution $\Delta t$ due to the time-of-flight depends on the broadening (uncertainty) of $T_0$, $T_s$, $t_t$ and $t_d$. In case of a moderated neutron beam, the broadening in time is dominated by the neutron transport in the target-moderator assembly, i.e. the component $t_t$. As discussed in Refs. [3,4], response functions for TOF measurements at a facility based on a spallation source are broader compared to those for measurements at a photonuclear source at the same distance. In addition, a more pronounced tail at long TOF values is observed. This difference is mainly due to the geometry of the target-moderator assembly, which is more compact for a neutron source based on photonuclear reactions.

Since the neutron flux decreases with increasing distance a compromise between resolution and intensity has to be made.

## 2.2. NRTA

In a transmission (or NRTA) measurement the observed quantity is the fraction of the neutron beam that traverses the sample without any interaction. For a parallel neutron beam which is perpendicular to a slab of material, this fraction or transmission T is given by:

$$\mathrm{T} = e^{-\sum_k n_k \bar{\sigma}_{tot,k}} \quad (4)$$

where $\bar{\sigma}_{tot,k}$ is the Doppler broadened total cross section and $n_k$ is the number of atoms per unit area of nuclide k.

Experimentally the transmission $T_{exp}$ is obtained from the ratio of the counts of a sample-in measurement $C_{in}$ and a sample-out measurement $C_{out}$, after subtraction of the background contributions $B_{in}$ and $B_{out}$, respectively:

$$T_{exp} = N_T \frac{C_{in} - B_{in}}{C_{out} - B_{out}} \quad (5)$$

where $N_T$ is a normalization factor accounting for the ratio of the total intensities of the incident neutron beam during the sample-out and sample-in cycles. The spectra in Eq. 5 are corrected for losses due to the dead time of the detector and electronics chain. The background is determined by an analytical expression applying the black resonance technique [4]. For this technique, samples of elements with strong absorption resonances (referred to as black resonance filters) are inserted into the beam. The free parameters in the analytical expression are determined by a least squares fit to saturated resonance dips observed in the TOF spectra, which result from measurements with black resonance filters. To account for the impact of the sample on the background, measurements are carried out with at least one fixed black resonance filter in the beam. A more detailed discussion on the background determination and the analytical expressions is given in Ref. [4].

Eq. 5 reveals that the experimental transmission is independent of both the detector efficiency and incoming neutron flux. Therefore, a transmission measurement can be considered as an absolute measurement which does not require additional calibration experiments or any reference to a standard

cross section [6]. In addition, the experimental observable $T_{exp}$ (Eq. 5) is a direct measure of the theoretical transmission (Eq. 4) if the measurements are performed in a good transmission geometry, which is [4]:

- the sample is perpendicular with respect to a parallel incoming neutron beam;
- all neutrons that are detected have passed through the sample; and
- neutrons scattered by the sample are not detected.

The conditions of an ideal or good transmission geometry can be achieved by a proper collimation of the neutron beam at the sample and detector position. However, it requires a homogeneous sample which does not contain holes. In case of inhomogeneous samples a special procedure is required, as discussed in Ref. [4,7].

## 2.3. NRCA

The observable in a capture (or NRCA) experiment is the fraction of the incident neutron beam undergoing a capture reaction in the sample. The theoretical capture yield $Y_\gamma$ resulting from a capture reaction can be expressed as a sum of primary $Y_{0,k}$ and multiple interaction events $Y_{m,k}$:

$$Y_\gamma = \sum_k \left(Y_{0,k} + Y_{m,k}\right) \quad (6).$$

The latter are due to a capture reaction after at least one neutron scattering event in the sample. For a parallel uniform neutron beam and a homogeneous slab of material perpendicular to the beam, the primary capture yield $Y_{0,k}$ resulting from a capture reaction by nuclide k is given by:

$$Y_{0,k} = \left(1 - e^{-\sum_j n_j \bar{\sigma}_{tot,j}}\right) \frac{n_k \bar{\sigma}_{\gamma,k}}{\sum_j n_j \bar{\sigma}_{tot,j}} \quad (7)$$

where $\bar{\sigma}_{\gamma,k}$ is the Doppler broadened capture cross section. Only in case of very thin samples and/or small cross sections, the capture yield is directly proportional to the product of the areal density $n_k$ and capture cross section. For relative thick samples, multiple interaction events have a substantial contribution to the yield and complicate the analysis as demonstrated in Refs. [2,3].

In a capture (or NRCA) experiment the prompt γ-rays, which are emitted after a neutron capture reaction in the sample are detected. The experimental quantity, which can be obtained from such an experiment and related to the theoretical capture yield, is the experimental yield $Y_{exp}$. This yield is derived from:

$$Y_{exp} = \frac{C_\gamma - B_\gamma}{\epsilon\, \Omega\, P_\gamma\, A\, \varphi} \quad (8)$$

where $C_\gamma$ and $B_\gamma$ are the observed dead time corrected sample and background spectra, respectively; φ is the incident neutron flux; A is the effective area of the sample seen by the neutron beam; P is the probability that the prompt γ-rays escape from the sample; Ω is the solid angle between sample and detector and ε is the probability to detect at least one γ-ray created in the capture event. To estimate the background, additional measurements without a sample in the beam and with a pure scattering sample (e.g. a carbon or $^{208}$Pb sample, which have a low capture cross section) are performed. A detailed discussion on the background determination is given in Ref. [4].

Eq. 8 reveals that the experimental observable in a NRCA experiment is much more complicated compared to the one obtained from a NRTA experiment. The yield $Y_{exp}$ can only be derived from the observed response once the incoming neutron flux and quantities which are related to the detection of the prompt γ-rays are known. Moreover, in most cases only the solid angle and effective area are independent of the energy of the incident neutron. The energy dependent neutron flux can be determined by measurement of a neutron standard reaction [4,6]. The efficiency to detect at least one γ-ray depends on the technique that is applied to measure the prompt γ-rays. Ideally, a detection system is used with an efficiency that is independent of the γ-ray cascade, i.e. independent of multiplicity and energy spectrum. Such a system can be realized by a total absorption detector with an almost 100 % efficiency or by applying the total energy detection principle, so that the detection efficiency becomes proportional to the total γ-ray energy produced in the capture event [8]. More details about such systems can be found in Ref. [4].

## 2.4. Data analysis

The areal densities of the nuclides present in the sample can be derived by a least squares adjustment, that is by minimizing the expression [4]:

$$\chi^2 = \left(Z_{exp}(t_m) - Z_M(t_m, \vec{\theta})\right)^T V_{Z_{exp}}^{-1} \left(Z_{exp}(t_m) - Z_M(t_m, \vec{\theta})\right) \quad (9)$$

where $Z_M(t_m, \vec{\theta})$ is a model describing the experimental observable $Z_{exp}(t_m)$. The theoretical estimate is the result of a folding to account for the response function $R(t_m, E)$ of the TOF spectrometer:

$$Z_M(t_m, \vec{\theta}) = \frac{\int R(t_m, E)\, Z'(E, \vec{\theta}) dE}{\int R(t_m, E) dE} \quad (10)$$

The theoretical model $Z_M(t_m, \vec{\theta})$ depends on parameters $\vec{\theta} = (\vec{\eta}, \vec{\kappa})$, which is a combination of resonance parameters and experimental parameters, represented by the vectors $\vec{\eta}$ and $\vec{\kappa}$, respectively. The resonance parameters $\vec{\eta}$ are used to parameterize the cross sections by the R-matrix theory. The experimental parameters $\vec{\kappa}$ include e.g. the detector and sample characteristics including sample temperature and the areal densities of the nuclides present in the sample.

The least squares adjustment can be performed by a resonance shape analysis (RSA) code, such as REFIT [9]. This code, which has been developed to parameterize cross section data in terms of resonance parameters, is based on the Reich-Moore approximation [10] of the R-Matrix formalism. It accounts for various experimental effects such as sample inhomogeneities, self-shielding and multiple interaction events, Doppler broadening, and the response of the TOF spectrometer and detectors. For the analysis of capture data special modules are included to correct for γ-ray attenuation in the sample and for the neutron sensitivity of capture detection systems [4]. Examples of the use of REFIT for NRTA and NRCA are given in Refs. [3,11,12].

## 3. NRTA and NRCA at GELINA

The Geel Electron LINear Accelerator (GELINA) of the EC-JRC-IRMM offers a pulsed white neutron source with an energy range from 10 meV to 20 MeV. A detailed description of the accelerator and its neutron producing target can be found in Ref. [13]. The main unit is a linear electron accelerator delivering very short electron pulses with energies up to 150 MeV and a maximum repetition frequency of 800 Hz. Electron bunches, with peak currents of 12 A in a 10 ns time interval, are compressed by a post-acceleration compression magnet to a width of less than 1 ns. The high-energy electrons generate Bremsstrahlung in a mercury-cooled rotating uranium target, where neutrons are produced by (γ,n) and (γ,f) reactions. To produce a neutron spectrum in the low-energy region, neutrons are slowed down in two 4-cm thick Be-containers filled with water and positioned above and below the uranium disk. Using suitable shielding materials in the target room, either the direct (fast) neutron spectrum with good time resolution may be used, or the moderated (slow) neutron spectrum. Most of the NRTA and NRCA measurements at GELINA are performed with a moderated neutron beam. To reduce the γ-ray flash and the fast neutron component a Cu and Pb shadow bar is placed close to the uranium target. The total neutron intensity is monitored by two $BF_3$ proportional counters located in the concrete ceiling of the target hall.

Transmission experiments in good transmission geometry can be performed at a 25 m and a 50 m measurement station using Li-glass scintillators as neutron detectors. The samples are placed in an automatic sample changer, which is positioned half-way between the detector and neutron producing target. Mostly measurements are performed with the accelerator operated at 800 Hz and a $^{10}B$ overlap filter is used to eliminate neutrons from a previous burst. At 25 m also measurement with a low operating frequency (100 Hz or 400 Hz) and a Cd overlap filter are carried out to investigate materials using low-energy resonances. At 25 m a NE905 Li-glass scintillator enriched to 95% in $^6Li$ is used. The scintillator (110.0 mm diameter and 12.7 mm thick) is placed in a thin-walled aluminium can and viewed by EMI9823-QKB photomultipliers (PMT). The detector at the 50 m station is a 6.35-mm thick and 101.6-mm diameter NE912 Li-glass scintillator, which is enriched to 95 % in $^6Li$ and viewed by one PMT.

Three measurement stations at 12.5 m, 30 m and 60 m distance from the neutron target can be used for the characterization of materials by NRCA. Depending on the elements of interest the accelerator is operated at 100 Hz, 400 Hz and 800 Hz using a Cd or $^{10}B$ overlap filter. The moderated neutron beam at the stations is collimated to about 75 mm diameter at the sample position. The detection system (i.e. γ-ray detectors, neutron flux detector, electronics and data acquisition system) are similar. The γ-rays are detected by $C_6D_6$ detectors. The energy dependent neutron spectrum is measured in parallel with a $^{10}B$ Frisch-gridded ionization chamber placed at about 80 cm before the sample. A double chamber is used with a cathode loaded with two back-to-back

layers of $^{10}$B. The chambers are operated with a continuous flow of a mixture of argon (90%) and methane (10 %) at atmospheric pressure.

NRTA and NRCA have been applied at the time-of-flight facility GELINA of the EC-JRC-IRMM for the characterization of reference samples for neutron induced reaction cross section measurements [11,12] and to study objects of cultural heritage interest [14-17]. Most of the archaeological applications so far are related to copper-alloy artefacts. Apart from Cu, they contain Sn or Zn as other major elements, and As, Ag, Sb, Co, Fe and In as minor or trace elements. In the course of several years NRCA has been extensively exploited to study various bronze objects of different origin: Etruscan statuettes [14], prehistoric bronze axes [15], Bronze-Age swords [16] and Roman metal objects like parts of water taps [17].

By using a position sensitive neutron detector (PSND) NRTA can be extended to have imaging capabilities. Therefore, a pixelated PSND was developed at the Rutherford Appleton Laboratory (UK) [18]. The PSND consists of 100 Li-glass crystals arranged in a 10 x 10 array. Each pixel is embedded in a support made of boron nitrate and coupled via a 0.5 mm thick glass disperser to a bundle of four 1-mm diameter acrylic optical fibres which transport the light to a 16-channel PMT. The prototype detector was characterized at GELINA [4,19]. When performing Neutron Resonance Transmission Imaging (NRTI) with a PSND, the ideal transmission geometry can not be fulfilled. Therefore special data reduction procedures have been defined based on measurements at GELINA. They are needed to assess the contribution of neutrons scattered in the sample to the background, as discussed in Ref. [19]. Examples of imaging measurements carried out at the ISIS facility are given Refs. [4,19].

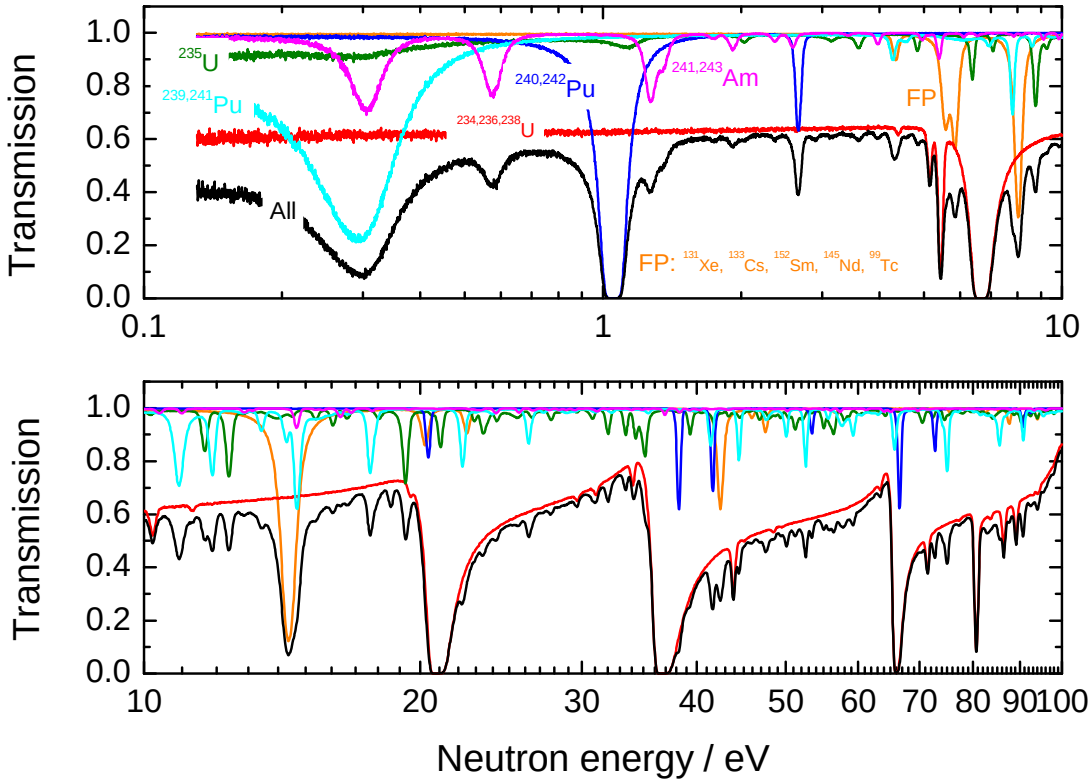


**Figure 2:** Transmission through a 1 cm thick sample with a composition that is similar to spent fuel. The contribution of the different nuclides present is illustrated by plotting separately the transmission due to the presence of only the fission products (FP); $^{241,243}$Am; $^{234,236,238}$U; $^{235}$U; $^{240,242}$Pu; and $^{239,241}$Pu.

## 4. Characterization of melted fuel by Neutron Resonance Densitometry

Neutron Resonance Densitometry (NRD) is being investigated as a method to quantify special nuclear material (SNM) in particle-like debris of melted fuel formed in severe nuclear accidents [20]. Characterization of such debris will be required for safeguards material accountancy at the time of removal of melted fuel resulting from the accident at the Fukushima Daiichi nuclear power plants. NRD relies on the use of NRTA together with NRCA combined with PGAA. The quantification of plutonium and uranium will rely on NRTA. NRCA combined with PGAA, using a well-type $LaBr_3$ detector, will be applied to determine the presence of impurities. In contrast to fresh or spent fuel, information about the elemental and isotopic composition of melted fuel formed after a severe nuclear accident is rather scarce. It is expected that the melted fuel will contain water, boron, concrete and structural materials. However, the corresponding elemental (and isotopic) composition cannot be predicted. The detection of specific prompt γ-rays can be used to identify the presence of nuclides which do not have resonances in the low-energy region (e.g. $^{10}$B, $^{28}$Si, $^{56}$Fe, $^{53}$Cr and $^{58}$Ni) and cannot be identified by NRTA.

The potential of NRTA for the characterization of fresh and spent fuel pins has already been demonstrated in Refs. [21, 22]. However, an accurate quantification of the amount of SNM in debris of melted fuel is much more complex. It requires a good understanding of the measurement process such that all components affecting the data are identified. The main sources of systematic effects are related to the specific characteristics of the samples, in particular, the sample inhomogeneity, particle size distribution, presence of neutron absorbing impurities, the total radioactivity and sample temperature. The complexity of the problem is illustrated in Figures 2-6.

In Fig. 2 the theoretical transmission as a function of neutron energy through a 1-cm thick sample is given. The composition of the sample, which is typical for spent fuel, was taken from Ref. [22]. The transmission is calculated for an ideal transmission geometry with the sample at 400 K and the detector paced at 10 m distance from the source. The contribution of the different nuclides present in the sample is illustrated by plotting separately the transmission due to the presence of only the fission products (i.e. $^{99}$Tc, $^{131}$Xe, $^{133}$Cs, $^{145}$Nd, $^{152}$Sm); $^{241,243}$Am; $^{234,236,238}$U; $^{235}$U; $^{240,242}$Pu; and $^{239,241}$Pu. This figure reveals that the determination of the amount of the fissile material is hampered by overlapping resonances due to the presence of the fission products, Am and the even U- and Pu-isotopes. Hence, the accuracy of the amount of fissile material will depend on the quality of the resonance parameters of the overlapping resonances. The dips corresponding to the strong s-wave resonances of $^{238}$U can be used to monitor the background level. However, additional studies are required to verify if the wings of transmission profiles of saturated resonances can be used to extract information about the $^{238}$U content.

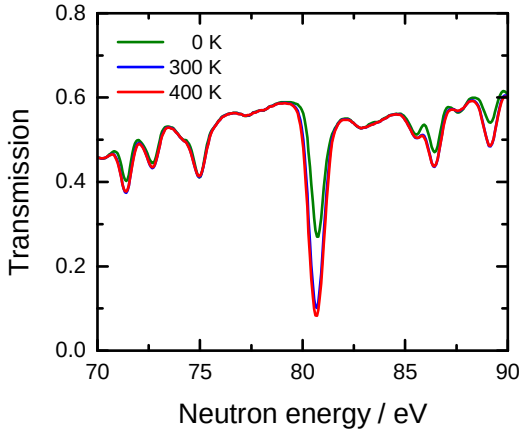


**Figure 3:** Transmission through a 1 cm thick sample with a composition that is similar to spent fuel. The transmission is given for the sample at 0K, 300 K and 400 K.

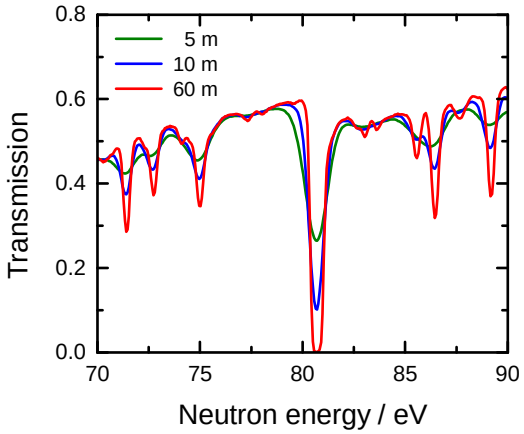


**Figure 4:** Transmission through a 1 cm thick sample with a composition that is similar to spent fuel. The transmission is given for the detector at a distance of 5 m, 10 m and 60 m.

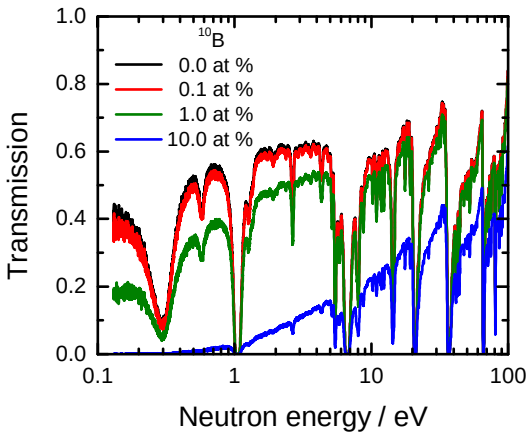


**Figure 5:** Transmission through a 1 cm thick sample with a composition that is similar to spent fuel and with different amounts of $^{10}$B (0, 0.1, 1 and 10 wt%).

For a correct quantitative interpretation of the data, the broadening of observed profiles due to both the Doppler effect and the response of the TOF spectrometer has to be taken into account. This is illustrated in Fig. 3 where the transmissions are shown for the sample at 0 K, 300 K and 400 K and in Fig.4 where the transmissions recorded with a detector at 5 m, 10 m and 60 m distance from the neutron source are compared.

Due to the large absorption cross section of the $^{10}$B(n,α) reaction in the low-energy region, the presence of boron will strongly influence the observed transmission. The attenuation of the neutron beam due to the presence of $^{10}$B is illustrated in Fig. 5 for different relative amounts of $^{10}$B in the sample. This additional attenuation does not only affect the base-line but also the area of the resonance dip. Therefore, it has to be taken into account in the analysis. Unfortunately, light elements such as $^{10}$B do not have resonances in the low-energy and special procedures are required to account for the presence of light matrix material.

One of the main difficulties for a correct interpretation of the transmission is the variety in size and shape of the particle-like debris. This can be concluded from results of transmission measurements using a $PuO_2$ pressed powder sample enriched to 99.93 wt% in $^{242}$Pu that was mixed with carbon powder and canned in a cupper container [7]. The measurements were performed at GELINA with the sample at 77 K and 300 K. In Fig. 6 the experimental transmissions are shown together with the results of a resonance analysis. In the analysis the areal density was adjusted and the resonance parameters were fixed to those

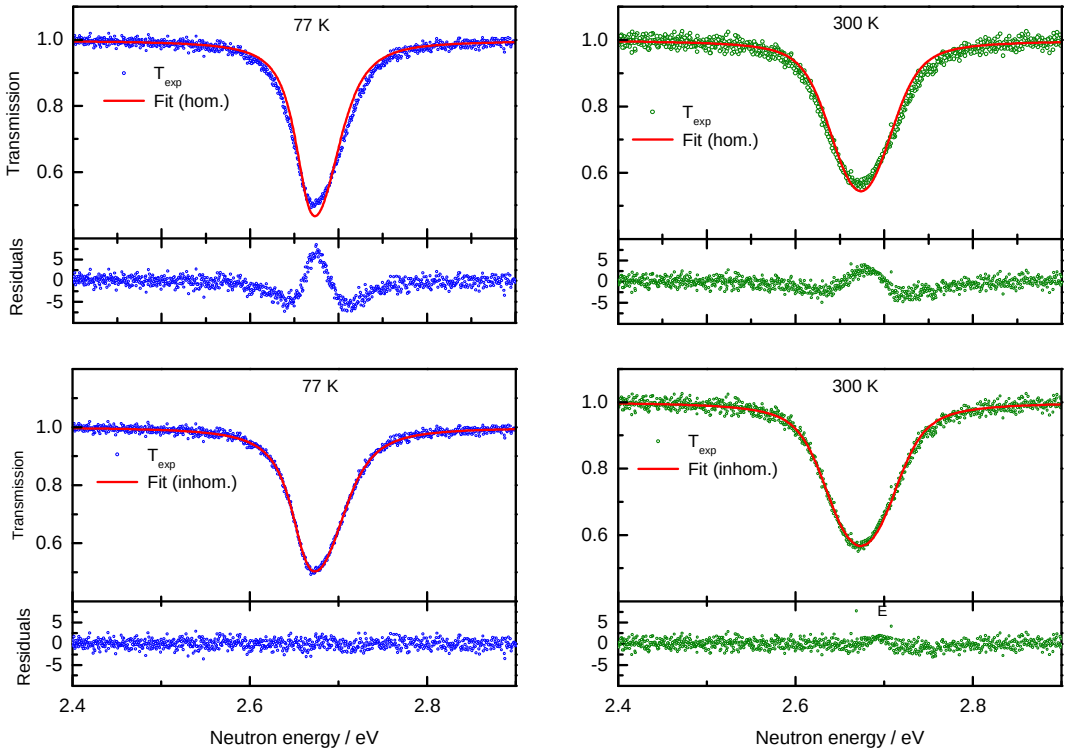


**Figure 6:** Transmission as a function of neutron energy for a $PuO_2$ powder sample enriched in $^{242}$Pu and mixed with carbon powder. The transmission is given for the sample at 77 K and 300 K. Results of a fit to the data to determine the areal density of $^{242}$Pu are given in case of a homogeneous sample and of a inhomogeneous sample.

recommended in the JEFF 3.1.2 evaluated data library. The areal density was derived in a fit to the experimental data supposing that the sample was completely homogeneous and by introducing an empirical macroscopic model that accounts for the sample inhomogeneities. This model, proposed by Kopecky et al. [7], includes a log-normal distribution describing the variation in areal density and a parameter reflecting the fraction of holes in the sample. The results are given in Table 1 and should be compared with the areal density 2.51 $10^{-5}$ at/b that was derived from weighing and a measurement of the area of the pressed pellet. The data in Table 1 show that there is a strong difference between the areal density which is derived from a fit with and without accounting for the sample inhomogeneities. Supposing an homogeneous sample the areal density resulting from NRTA is biased to lower values. The values at 77 K and 300 K derived by including an areal density and a holes fraction are fully consistent with 2.51 $10^{-5}$ at/b. In addition, the residuals in Fig. 6 demonstrate that the quality of the fit improves significantly when the powder grain size and fraction of holes in the sample are included. The improved quality is observed for both the 77 K and 300 K data.

| | Areal density $^{242}$Pu at/b | |
|---|---|---|
| | 77 K | 300 K |
| Homogeneous | 1.65 $10^{-5}$ | 1.79 $10^{-5}$ |
| Inhomogeneous | 2.49 $10^{-5}$ | 2.47 $10^{-5}$ |

**Table 1:** Areal density of $^{242}$Pu in a powder determined by NRTA for a homogeneous and inhomogeneous distribution of the areal density (uncertainty due to counting statistics: 4 $10^{-8}$ at/b)

## 5. NRD development at GELINA

To study the systematic effects mentioned in section 4 the Japan Atomic Energy Agency (JAEA) and the Joint Research Centre of the European Commission (EC - JRC) started a collaboration. In this collaboration various aspects in the development of NRD, from basic measurement principles up to detector and accelerator development, are included. One of the most important activities is the study of the influence of the sample characteristics in order to improve the analysis procedures and to define target values with realistic uncertainties. This study includes the development of theoretical models to account for sample inhomogeneities and strongly relies on measurements performed at GELINA. The experimental data will be used to validate theoretical models, to improve relevant nuclear data if needed, to determine reference spectra and to define performance values. In addition, the resonance shape analysis code REFIT [9] will be modified and adapted to the needs of NRD.

Several measurement campaigns are scheduled at GELINA using samples which are dedicated to a specific part of the problem. Measurements on pure Cu metal discs with different thicknesses at the 25 m station will be performed to study the accuracy of NRTA in case of homogeneous samples. The results of these measurements will be used to study the influence of the resonance characteristics (i.e. resonance strength) and to verify if from the wings of the transmission profiles of saturated resonance reliable information can be extracted. In addition, measurements with boron and lithium samples added to the Cu discs will be carried out to define a method to account for the presence of matrix material which does not have resonances in the low-energy region. These measurements have already been completed and the analysis of the data is in progress. Parts of the data are shown in Fig. 7 and 8.

In another contribution to this ESARDA symposium different models to account for the density variation in powder samples are described and calculated [23]. The performance of the models has been compared using results of stochastic numerical simulations as a reference. This comparison will be complemented with experimental data. The data result from transmission measurements using pure metal Cu and W samples and samples made out of Cu and W powder mixed with S powder. The experiments are in progress and will be finished by July 2013.

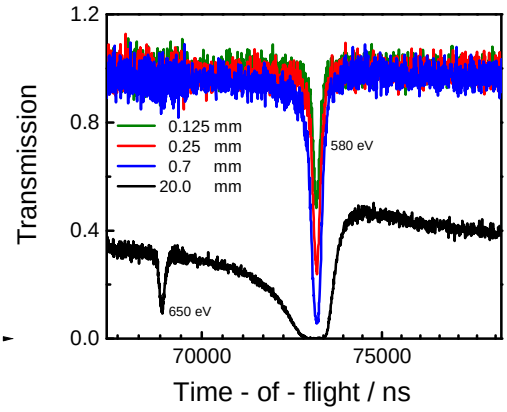


**Figure 7:** Experimental transmission around the 580 eV resonance through Cu discs with different thickness.

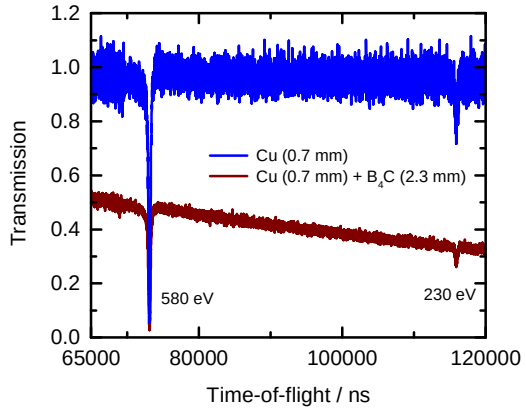


**Figure 8:** Experimental transmission around the 580 eV and 230 eV resonance through a 0.7 mm thick Cu disc with and without a 2.3 mm thick $B_4C$ disc.

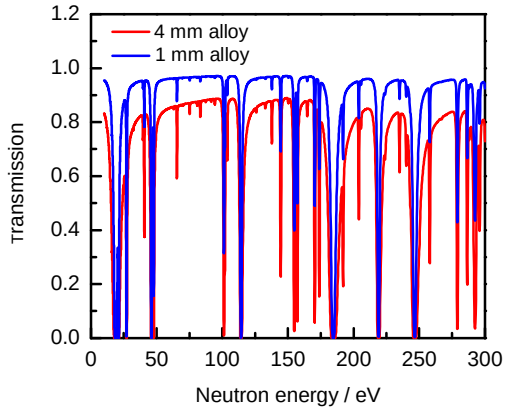


**Figure 9:** Theoretical transmission through a W-In-Zr-Fe-Cr alloy sample (1 mm and 4 mm thick).

| Element | at. % | wt. % |
|---|---|---|
| W | 47.0 | 64.84 |
| Zr | 47.0 | 32.17 |
| In | 1.2 | 1.03 |
| Fe | 3.0 | 1.26 |
| Cr | 1.8 | 0.70 |

**Table 2:** Elemental composition of the alloy used as non-radioactive alternative for a melted fuel sample.

For a final validation of the models an alloy consisting of W, In, Zr, Fe and Cr has been defined and is being produced. The elemental composition of the alloy is given in Table 2. Considering the resonance characteristics and material properties, W and In have been chosen to replace the role of U and Pu, respectively. Two homogeneous discs with a nominal thickness of 1 and 4 mm will be made. In addition, the remaining cast material will be used to produce samples that are similar in shape as the debris of the melted fuel. A simulation of the transmission through the 1 mm and 4 mm homogeneous disc is shown in Fig. 9. The transmission measurements will be performed at different sample temperatures.

## 6. Summary

A method, referred to as Neutron Resonance Densitometry, has been presented. The method, which relies on the appearance of resonance structures in neutron induced reaction cross section, is being developed for the characterization of melted fuel that is formed after a severe nuclear accident. The basic principles have been explained and special problems related to measurements of particle-like debris have been presented. In addition, the programme to study these problems, to develop dedicated analysis procedures and to assess the performance of NRD has been discussed. This R&D programme, which is part of a collaboration between JAEA and EURATOM, strongly relies on measurements at the time-of-flight facility GELINA installed at the EC-JRC-IRMM.

## 7. Acknowledgements

This work is part of a collaboration between the Japan Atomic Energy Agency (JAEA) and European Atomic Energy Community (EURATOM) represented by the European Commission in the field of nuclear materials safeguards research and development.

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# Safeguards implementation and status of Posiva's encapsulation plant and geological repository

**Mari Lahti**

Posiva Oy
Olkiluoto
FI-27160 Eurajoki, Finland
E-mail: mari.lahti@posiva.fi

***Abstract***

*The licensing process of the Finnish final repository has advanced into the submission of the construction license application in the end of 2012. Posiva's final disposal solution is based on a facility complex consisting of an above ground encapsulation plant and an underground disposal facility that are connected by a vertical shaft that transfers the disposal canisters into the repository. The construction of the encapsulation plant and further underground rooms needed for the disposal facility are expected to start in 2015. Safeguards implementation during the construction and operation of the nuclear facilities has been outlined in the construction license application. In the same context Posiva also submits the basic technical characteristics for the two facilities.*

*Posiva's operations from the start of the construction of the underground rock characterisation facility ONKALO in 2004 have been under the national safeguards control. The control ensures that ONKALO is excavated as declared and that any other excavations compromising the integrity of the geological formation do not exist in the area. The safeguards control will be continuous to cover the whole period from the rock characterisation phase to the licensing and construction of the nuclear facilities. The forthcoming excavations of the deposition tunnels will be controlled based on the experience gathered during the ONKALO phase. The safeguards implementation at the encapsulation plant and disposal facility will expand to include the nuclear material accountancy and reporting together with the control of the nuclear material pathways and storage. The accountancy and reporting will be integrated into the spent nuclear fuel database developed during the coming years before the commissioning phase. The development targets also comprehend the unique identification for the disposal canisters. Moreover, Posiva prepares to provide positions for the verification measurements of the fuel assemblies before the encapsulation.*

**Keywords:** repository; encapsulation; licensing; safeguards

## 1. Introduction

Posiva was established in 1995 to be an expert organisation responsible for the final disposal of spent nuclear fuel of its owners. Posiva is responsible for the research related to the geological final disposal of spent nuclear fuel and for the construction, operation and eventual decommissioning and dismantling of the encapsulation plant and disposal facility after about hundred years of operation.

Posiva's final disposal plan is based on the KBS-3 concept developed by the Swedish Nuclear Fuel and Waste Management Company, SKB. The concept is based on the multiple release barrier system, where the radioactive material is isolated from the organic nature. The release barriers include the physical state of the fuel, the disposal canister, the bentonite buffer, the backfilling of the tunnels and the surrounding host rock. The fuel assemblies will be mounted into the copper and cast iron canister that is corrosion resistive and protects the fuel from the mechanical strain. The canisters are emplaced into several hundred meters depth into the bedrock and they are enclosed by bentonite. The bentonite will protect the canister in case of tectonic movements in the bedrock and lower the groundwater flow in the vicinity of the canister. The bedrock provides stable and predictable environment for the bentonite and the canisters. The deep emplacement will

isolate the canisters from the future climate change and the human accessibility.

## 2. Licensing process

Posiva has applied in 1999, 2000 and 2008 Governments Decision in Principle that the planned construction of the encapsulation plant and geological repository for the final disposal of spent nuclear fuel at Olkiluoto in Eurajoki is in the public interest. The Governments decisions were ratified by the Parliament in 2001, 2002 and 2010 accordingly. The valid Decisions in Principle cover the 9000 tU that are and will be produced in two units in Loviisa and in four units at Olkiluoto.

Posiva submitted the construction licence application for the encapsulation plant and disposal facility in the end of 2012. Connected to the handling of the application, the Finnish Radiation and Nuclear Safety Authority (STUK) will make a safety assessment for the Ministry of Employment and the Economy. The safety assessment will pay special attention to the long-term safety and technical feasibility of the facilities and the operational safety of the encapsulation plant. STUK will also examine the plans for nuclear material safeguards and the arrangements for physical protection. The safety assessment may influence the final design and layout of the facilities and the concept.

The license handling time is expected to be about two years. The construction of the nuclear facilities would start in 2015 correspondingly. During the handling time Posiva will continue the demonstrations in the underground rock characterisation facility ONKALO and prepare for the construction of the nuclear facilities. Posiva is planning to submit the operation license application in 2020.

## 3. Technical characteristics of the facility complex

The construction license application presents Posiva's design for the facility complex consisting of the above ground encapsulation plant and the underground disposal facility (Figure 1). The facilities are connected by a vertical shaft that transfers the disposal canisters into the repository. Also, the safeguards implementation during the construction and operation of the nuclear facilities has been outlined in the application. In the same context Posiva submits the basic technical characteristics for the two facilities. The BTC data provided describe the design of the facilities as well as the flow of spent nuclear fuel from the encapsulation into the geological repository.

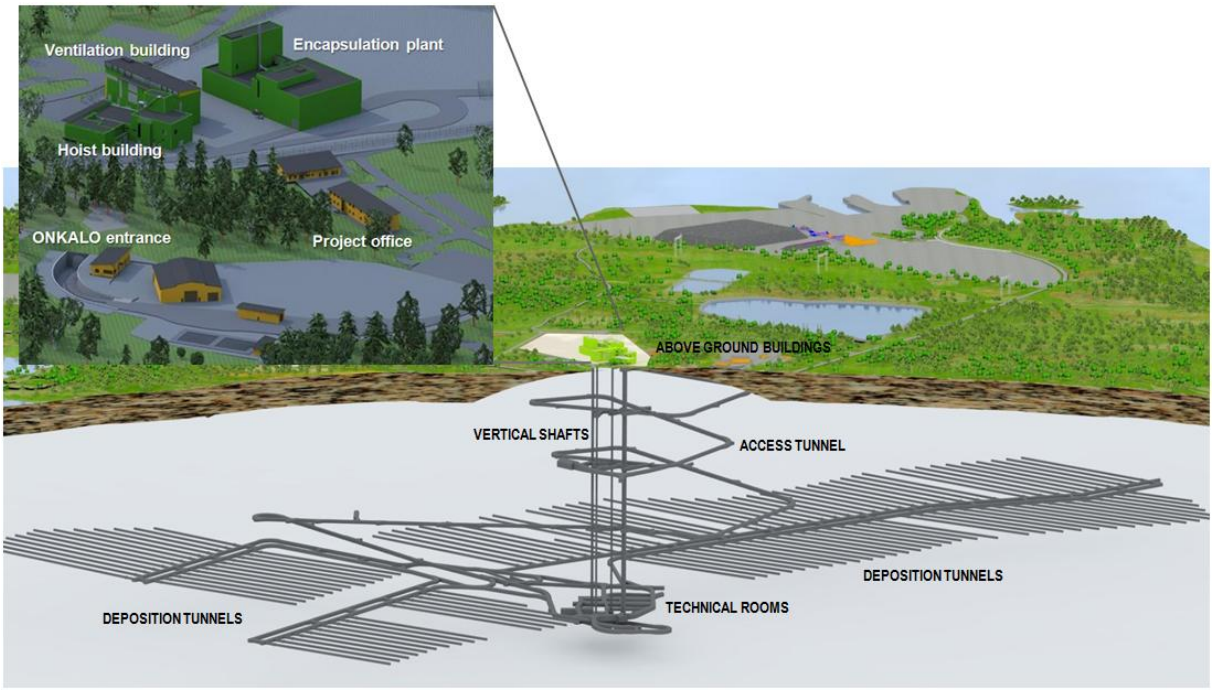


**Figure 1.** Posiva's design for the facility complex consisting of the above ground encapsulation plant and the mainly underground disposal facility.

### 3.1. Encapsulation plant

The plant will encapsulate the spent nuclear fuel produced by the currently operating four NPP units (Loviisa 1 and 2, Olkiluoto 1 and 2), the Olkiluoto 3 under construction and the planned Olkiluoto 4. The encapsulation plant will receive the spent nuclear fuel assemblies packed into appropriate transport casks. The encapsulation process comprehends packing of the spent nuclear fuel assemblies into disposal canisters. The encapsulation plant will send the canisters into the disposal facility. The main areas of the encapsulation plant will be the receiving area, the fuel handling cell, the welding and weld inspection stations, the canister buffer storage and entrance to the canister lift (Figure 2). The plant also has storage rooms for copper canisters and bentonite blocks.

The copper canisters and the transport casks containing the spent nuclear fuel will arrive to the receiving area on the ground level of the encapsulation plant (level +10.30). The transport cask will be lifted from the receiving area into the transfer corridor by a crane and moved below the handling cell. The handling cell has a docking station, where the cask is opened and the fuel assemblies are unloaded. The fuel assemblies will be dried in the handling cell for removing any residual moisture after interim storing and transportation. In addition to drying the fuel assemblies will be identified and verified and emplaced one by one into the disposal canister. The copper lid of the canister will be sealed by electron beam or friction stir welding. The weld will be machined and inspected by four complementing methods for ensuring the quality of the welding. The methods are visual, eddy current, ultrasonic and radiographic inspections. The acceptance criteria for the weld will be based on a combination of inspections. Inspected and approved canisters will be transferred into the storage either at the encapsulation plant (level +1.90) or directly into the disposal facility (level -437).

### 3.2. Disposal facility

The disposal facility will receive the encapsulated spent nuclear fuel from the encapsulation plant to be finally disposed into the geological repository at about 400 m to 450 m depth. The disposal facility will consist of the above ground ventilation and hoist building and the underground parts such as the access tunnel, the technical rooms, the shafts for ventilation, personnel entrance and canister transfer, the central tunnels and the repository. The repository consisting of the deposition tunnels and deposition holes will be excavated after the construction license is granted. The preparatory phase prior to operation is expected to start in 2015 when the first deposition tunnels will be excavated (Figure 3). In the course of the disposal operation the disposal areas will be constructed in steps.

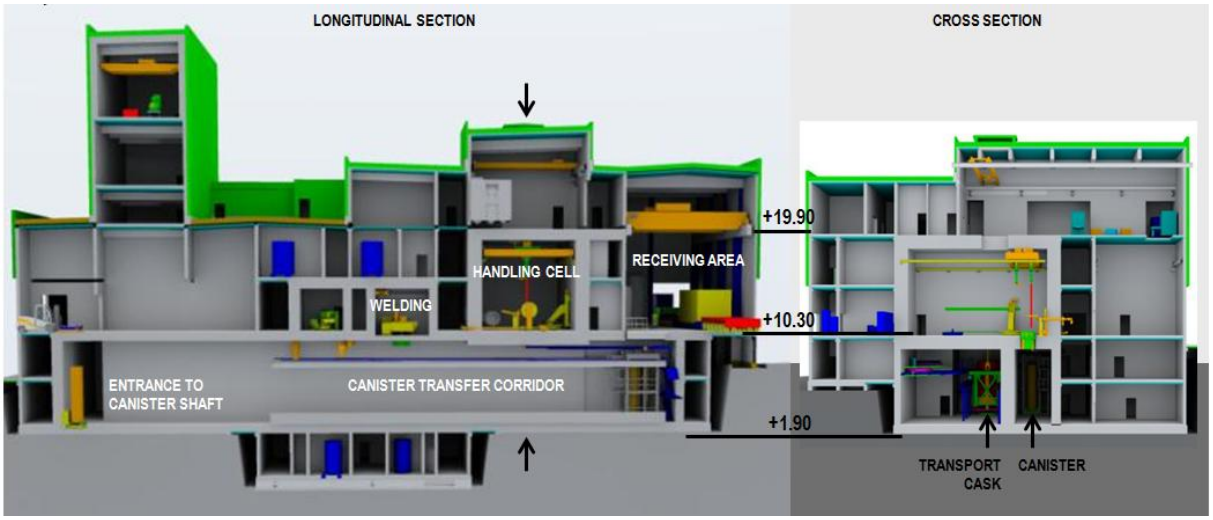


**Figure 2**. Longitudinal and cross sections of the encapsulation plant to show the main areas of the facility. Location of the cross section in relation to the longitudinal section is shown by arrows.

The disposal facility will have altogether six connections to the ground level. Four of the connections will be completed during the ONKALO phase (the access tunnel, personnel shaft and two ventilation shafts) while the rest will be finished before the disposal facility is commissioned. ONKALO and the disposal facility are designed so that ONKALO will serve as part of the disposal facility when disposal operations commence. The vehicle access tunnel is designed and dimensioned to allow transports during the disposal phase. The access tunnel will be used for transporting crushed rock and backfilling material. The above ground hoist building will serve as entrance to the disposal facility. The personnel traffic will be arranged mainly by the personnel hoist.

The disposal canisters will be transferred in the canister shaft by a lift. The canister reception area will be located in the technical rooms at the -437 m level. The canister storage is directly connected to the canister lift. The reception area has two floors, the canisters are unloaded and stored at the lower level and lifted to the upper level for the transfer to the deposition tunnels. The same lift will carry the bentonite blocks from the encapsulation plant to be unloaded at the upper level of the reception area. These blocks will be used for lining the deposition holes before emplacing the disposal canisters. Additionally, the operating waste produced at the encapsulation plant will be moved using the canister lift into the operating waste repository at the -180 level.

The transfers of disposal canisters, operating waste and bentonite blocks will be segregated. The canister lift will always carry only one type of material at a time and the transfers of each type can be done in campaigns. Also, the canister transfers will be segregated from blasting during the extension of the repository. The radiation monitoring can be used for verifying the transported material. The canister shaft will have connections to the access tunnel on several depth levels. Radiation monitoring and structural solutions will ensure that canisters are only carried to the deep repository level. Personnel will only be present for inspection and servicing of the canister lift.

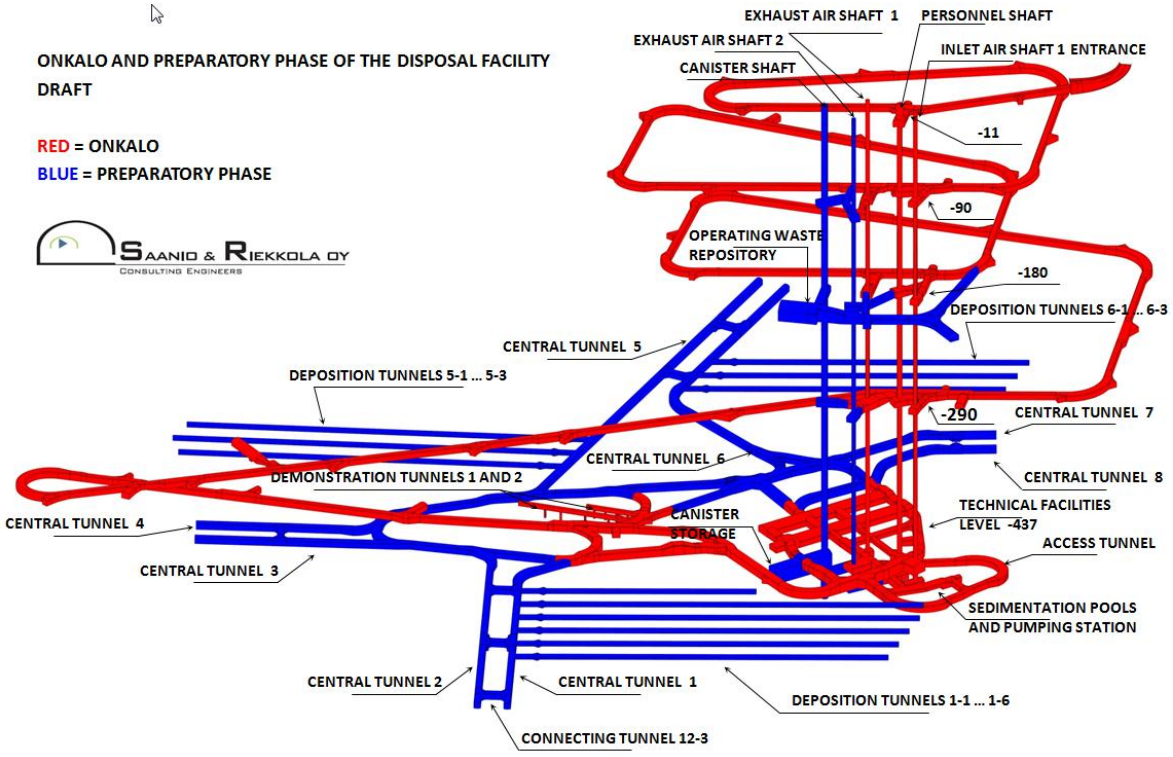


**Figure 3.** ONKALO and the preparatory phase of the disposal facility. The red part represents the ONKALO phase that is almost completed in 2013. The blue parts constitute the preparatory phase that will start around 2015 and is planned to be finished before the start of the operation.

### 3.3. Flow of spent nuclear fuel in the facilities

The design capacity of the encapsulation plant is 100 canisters annually, but in the early phase of the operation the expected encapsulation rate is 40 canisters per year. The schematic flow of the nuclear material through the facilities is presented in Figure 4. The main areas at the encapsulation plant, where spent nuclear fuel will be handled or stored, are the receiving area, the handling cell and the canister storage. The canister transfer corridor will be pathway from the handling cell into the canister storage and to the canister lift.

The receiving area is the only accessible storage room at the encapsulation plant although the fuel assemblies will be inside the closed transport cask and inaccessible. The receiving area has storage capacity for four loaded or empty casks. Transport casks can be opened and the fuel assemblies counted and identified only inside the handling cell. The handling cell will be inaccessible, but the identifying can be arranged by camera surveillance. In the handling cell the spent nuclear fuel assemblies are either in the transport cask, drying station or in the disposal canister. Positions may be temporarily closed depending on the process phase. The design basis is that all fuel assemblies shipped from the interim storage can be handled at the encapsulation plant and they match the ideal combination to fill the canisters. Situations when fuel assemblies would be shipped back to the interim storage are exceptional. Abnormal situations will be solved at the encapsulation plant as far as possible, for example leaking fuel assemblies will be handled and disposed if possible.

The fuel handling steps in the handling cell will be drying, verification and encapsulation. The disposal canister will be closed inside the handling cell but the mounting of the copper lid and the welding will be carried out in the welding station. During the welding and weld inspection the canister will be on the trolley in the transfer corridor and the upper part of the canister is lifted into the station. After welding and inspection the canister will be moved from the end of the corridor into the canister store or into the canister lift. The canister store at the encapsulation plant will contain up to 12 ready disposal canisters. The store will be inaccessible but monitored by cameras.

The inventory locations in the disposal facility are the underground canister store (capacity 30 canisters) and eventually the deposition holes. Both locations will be inaccessible although the canister store will be controlled by camera surveillance. The deposition tunnels will not contain constant camera surveillance. The transfer and installation vehicle carrying the disposal canisters from the canister store into the deposition holes will have several cameras to assist the operations. The bentonite buffer blocks will be installed on top the canister immediately after the canister is mounted into the hole. The deposition tunnels will be accessible until the tunnel is backfilled and plugged, but the deposition holes will be protected by temporary covers.

Potential case of backward flow would be transferring damaged or inspection failed disposal canister back into the handling cell at the encapsulation plant for unloading. The welded canister can only be dismantled by machining the upper part. The machining can be carried out in the transfer corridor using the same equipment that machines the canister weld. Machining will destroy the weld and the engraved identification. The fuel assemblies would be unloaded in the handling cell and stored in the transport cask or drying station before loading them into a new disposal canister. Dismantled canisters will probably not be reused as they are but at least the insert can be utilized if it is not damaged. The canister is not expected to be contaminated.

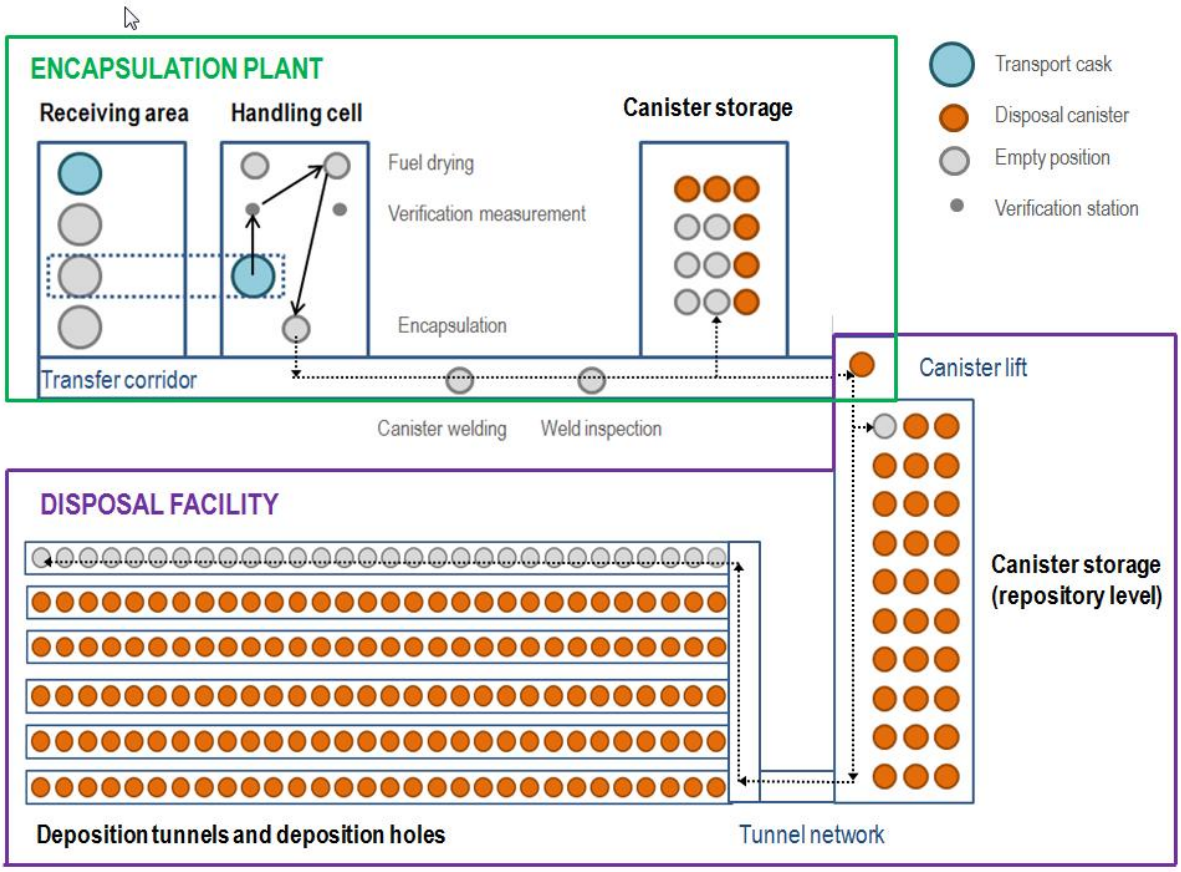


**Figure 4.** The schematic flow of spent nuclear fuel through the facilities.

## 4. Development of the safeguards control

Posiva started the construction of the underground rock characterisation facility ONKALO in 2004. STUK has implemented safeguards control for the whole ONKALO phase. Posiva has compiled a handbook for instructing the safeguards reporting and control. The handbook has been approved by STUK originally in 2005 and it has been updated several times to fulfil the developing needs. STUK has performed on average three annual periodic inspections to ensure that ONKALO has been constructed as declared. The safeguards reporting comprise annual reports and three periodic reports per year that describe the advancing of the construction and absence of undeclared activities. The reports consist of as built drawings, laser scanning images and results of the microseismic monitoring.

The safeguards control concepts for the geological final disposal are currently developed. Posiva has described the safeguards arrangements during the construction and operation of the nuclear facilities in the non-proliferation plan attached to the construction licence application. The safeguards control developed during the ONKALO phase will be updated to cover the preparatory and operational phase. The nuclear material accounting and reporting system will be connected to the spent nuclear fuel database. The methodology for the unique identification of the disposal canisters and monitoring of the nuclear material pathways and stores will be developed in the coming years. Before encapsulation the fuel assemblies will be verified to ensure the quality and quantity of the nuclear material that will be finally disposed and beyond traditional safeguards control. Posiva is preparing to provide a position for the verification measurement in the fuel handling cell at the encapsulation plant. The aim is that in the commissioning phase Posiva has a valid nuclear material handbook that covers all the aspects of the national and international safeguards control needed for the final disposal.

# Concepts for dismantlement verification and neutron multiplicity measurements for plutonium mass attribute determination

**Malte Göttsche[1], Paolo Peerani[2], Gerald Kirchner[1]**

[1]University of Hamburg
Centre for Science and Peace Research
Beim Schlump 83, 20144 Hamburg, Germany
E-mail: malte.goettsche@physik.uni-hamburg.de

[2]Joint Research Centre
European Commission

***Abstract:***

*For meaningful and effective verification of nuclear warhead dismantlement, a combination of warhead authentication measures along with a robust chain-of-custody would be instrumental. In this context, requirements and concepts will be discussed that could tie together warhead authentication and chain-of-custody measures to a meaningful overall regime. For this purpose, it will be assessed at what stages information barriers and other technologies could be applied. This serves as an overall introduction into the topic of dismantlement verification.*

*A more technical part focuses on passive neutron multiplicity measurements with He-3 detectors for the purpose of verifying attributes that are helpful for authenticating warheads. With a given isotopic composition (e.g. deduced from gamma spectroscopy measurements), neutron multiplicity counting is suited for determining plutonium mass which is considered one attribute. Plutonium measurements have been performed in the PERLA laboratory in Ispra that were used to validate MCNPX PoliMi simulations. Preliminary simulation results will be presented.*

**Keywords:** disarmament, verification, information barrier, neutron multiplicity counting

## 1. Arms Control Verification

As part of verified fissile warhead component inventory declarations and verified warhead dismantlement, two elements will most likely be key to a sound verification regime. On the one hand, this is the authentication of warhead (components). Authentication in this context is the process during an on-site inspection through which it is assessed by measurements whether a specific item is a nuclear warhead (or component). On the other hand, a robust Continuity of Knowledge, the other key element, could be defined as providing means to effectively demonstrate over a certain time or process, e.g. during warhead dismantlement, the unchanged identity of the treaty-accountable item (i.e. that it remains the same item) and the integrity (i.e. that no undeclared changes to the item occurred). Technologies relevant for providing a Continuity of Knowledge are for example seals and tags that can be applied on the items directly or their storage containers and monitoring equipment.

Due to the classified nature of the items under investigation, direct measurements for authentication purposes will most likely not be possible as they would reveal information that is considered sensitive for nonproliferation, national security and possibly other reasons. The use of information barriers could overcome this problem. An information barrier takes classified measurements but converts the results to an unclassified output (such as a binary yes/no signal) while protecting the sensitive data from the inspector's view. The range of possible measurement techniques includes some non-nuclear type measurements and gamma spectroscopy, though this paper will focus on neutron detection.

Recognizing that other reasonable approaches exist, issues relating to the attribute approach are considered here. In the attribute approach, the inspecting and host parties agree on a set of attributes that the items would be checked against and on an analysis algorithm. This set should be defined in a way that it allows for an assessment whether a declared warhead component is genuine. One of the attributes that could be considered is a threshold fissile mass.

The technical focus of this paper will be on authentication of dismantled fissile warhead components that would be stored in appropriate containers with the attention being directed towards passive neutron multiplicity measurements for determination of plutonium fissile mass. Neutron interactions will be shortly introduced, then plausible container configurations will be described to then analyze what physical interactions occur in the containers that have an influence on the analysis. This will be done by simulations that are also checked against experimental results.

Neutron Multiplicity Counting

The neutron flux emitted by a fissile sample is affected by a number of possibly unknown properties [1]:

- spontaneous fission rate (the goal of neutron multiplicity counting to deduce fissile mass)
- sample self-multiplication / variation across the sample, in particular through induced fission
- $(\alpha, n)$ reaction rate if oxides are present

Other properties can be eliminated by careful calibration and counter design or are small or constant as described in [1]**.** Given that spontaneous fission, multiplication and $(\alpha, n)$ reactions all have an influence on neutron emissions, the goal is to determine the spontaneous fission rate since – through the known fission rate per Pu-240 atom – the fissile mass can be deduced if the isotopic composition of the sample is known through other means, e.g. gamma spectroscopy.
Passive neutron multiplicity counting can separate the contributions of these three effects to the overall neutron emission without requiring representative reference materials. Multiplicity counting measures the multiplicity distribution (i.e. neutron correlations) and calculates three parameters (Singles, Doubles and Triples rate) so that the three unknowns can be solved. Initial determination of detector parameters can be done with a Cf-252 source alone. Multiplicity counting based on this calibration can be slightly biased because of a detector's different efficiencies between Cf-252 and Pu fission neutrons [2], but this effect is minor.

## 2. Technical Implications of Component Storage Containers

The determination of fissile mass using neutron multiplicity counting requires information on isotopic composition that could come from gamma spectroscopy measurements. Gamma rays are shielded in particular by materials with a high atomic number. This paper only looks at the influence of neutron shielding but it is acknowledged that gamma shielding can complicate the quantitative analysis significantly as well.

Neutron interactions

Neutrons can react via elastic or inelastic scattering as well as neutron-induced nuclear reactions. Elastic scattering slows them down and changes their direction. The average energy loss due to elastic scattering is given as $2E \cdot A/(A+1)^2$ ,where A is the mass number of the nucleus and E the energy of the neutron [3]. It is clear that the energy loss decreases with increasing atomic number A. The probability of many neutron-induced reactions drops off rapidly with increasing neutron energy and is usually high for low energy neutrons.

Containers

The container's purpose is mainly to ensure safety and radiation protection. It appears that the main requirements for container certification are criticality control, energy absorption (shielding), thermal insulation and fire resistance [4, 5]. Also, containment and impact limitation (shock mitigation) must be guaranteed [5, 6]. A range of certified containers exist. They might contain some lead shielding which is very relevant to gamma radiation, but less to neutrons. However, usually a material like Celotex is used which consists of hydrogen, carbon and oxygen contents. Celotex has a density of 0.24 - 0.29 g/cm$^3$ [5]. While this material is suited for absorbing neutron energy via elastic collisions, it does not naturally contain significant neutron poison such as boron.

The container used for pit storage at PANTEX, the US site for warhead production and the only site where they are dismantled, which hosts a warhead component storage site, is the AL-R8 container which uses Celotex.[1] Other containers that are also certified are the 9975 and AT-400 container.[2] The 9975 container can be seen in Figure 1.

For the purpose of analyzing the physical processes that occur in the container, total neutron counting will be looked at, bearing in mind that those physical processes could have an influence on the neutron multiplicity counting measurements. In this study, the 9975 container will be investigated as it has the thickest neutron moderator compared to the AT-400 and AL-R8.

Code check and simulations

At the PERLA Laboratory of the Joint Research Centre site in Ispra, Italy, neutron measurements have been performed on a 2861 gram $PuO_2$ powder sample (70% Pu-239 and 24% Pu-240 and other plutonium isotopes) that was placed inside the 9975 container. A neutron detector system was designed to enclose the container. A model of the measurement setup has been built in MCNPX PoliMi. Experimental data will be compared to the simulated results to serve as a check of the simulation code and the implemented detector and container geometry.

The detector consisted of 8 slabs (4 lateral slabs, 2 back slabs and 2 front slabs. Each of these slabs contained six He-3 detector tubes (cylindrical tubes with a 2.44 cm diameter and 35.8 cm lengths). The slabs were made of polyethylene to moderate the neutrons coming in. The geometries of the slabs were not identical but similar in shape. They were about 23.5 cm broad and 9 cm long. The detector (MCNPX PoliMi model) is sketched in Figure 2.

The experiment yields the singles count rate S=81765±302. In the simulated result, we obtain S=83070±289. This shows that the simulated count rate overestimates the experimental by 1.6%±0.5%. All errors are statistical. The simulated lies outside the experimental result with its uncertainty, but can still be considered very small. Therefore this MCNPX PoliMi model seems to be consistent with the experiment in this regard and is used for further simulations.

Since the ultimate goal is to authenticate nuclear weapon components, we simulate weapons-usable fissile material. The sample simulated is a 2 kg solid sphere filled with 95% Pu-239 and 5% Pu-240 metal (19.8 g/cm$^3$). The goal here is to simulate a sample somewhat similar/comparable to a weapon component so that conclusions drawn from the simulations can in general terms hold for assessing the physical processes that would occur during verification.

[1] More information on the design of the AL-R8 is found in [5, 7, 8]
[2] More information on the design of the AT-400 is found in [7]. More information on the design of the 9975 container is found in [6, 9, 10].

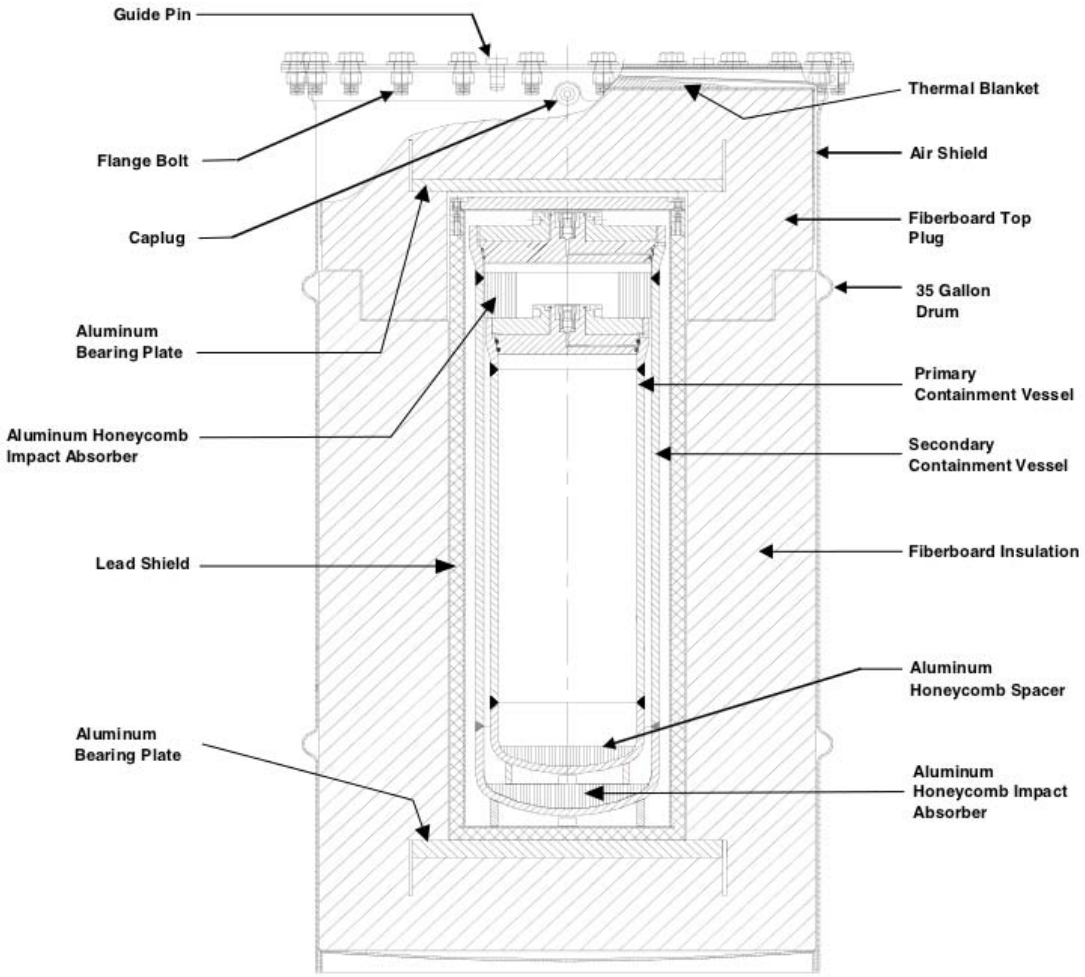


**Figure 1: Sketch of 9975 container from [9]. The fiberboard insulation consists of Celotex. The Celotex component is 11.9 cm thick, the overall container has a diameter of 46.4 cm.**

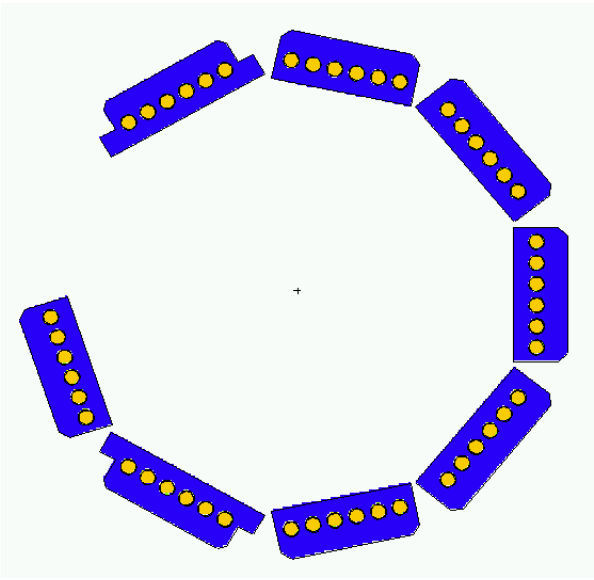

**Figure 2 Cross-sectional sketch of the He-3 slab detector from the top. Polyethylene is depicted in blue; the He-3 tubes are marked yellow. The tubes have a length of 35.8 cm, items with a diameter less than 35.8 cm can be placed in the detector.**

Regarding fissile material mass, 2 kg can be considered a reasonable fissile material quantity for a weapon, as for example discussed by Thomas Cochran in [11] who estimates that 2 kg weapon-grade plutonium is needed for a 10 kt yield weapon under a high technical capability. A solid sphere was chosen as opposed to a hollow sphere since the effect of multiplication becomes larger which is one of the physical processes that have an influence on and complicates the analysis of the measurement results. The rate of the neutrons emitted from spontaneous fission in this source is $1.02 \times 10^5$ n/s.

Simulations with the original 9975 container with Celotex are performed. For comparison, a simulation without the container was done. Furthermore, a simulation was run with 5 weight percent borated Celotex to study the effects of a neutron poison if it should be included in a container for whatever reasons.

Results

As can be seen from Table 1, the neutron detection events (through neutron capture) in the He-3 counter increase by 33% when the 9975 container is present. With 5% borated polyurethane, the count rate is still 22% above the rate without container, but 8% below that of the 9975 container with Celotex. Three different effects lead to this result and will be discussed, namely neutron moderation in the container, neutron capture in the container (with a negative effect on the count rate) and additional induced fission in the plutonium sample due to back-scattering from the container.

*Neutron moderation*

Celotex consists of low-Z materials that effectively reduce the neutron energies due to elastic scattering. This effect is in addition to elastic scattering that takes place in the detector polyethylene, so it further contributes to slowing down the neutrons to increase their detection probability in the He-3 detectors since the He-3 capture cross-section is high for thermalized neutrons, see Figure 4. This effect can be seen in Figure 3 which shows the significant increase of the neutron surface current in the thermal region. Also the configuration without container has a significant thermal contribution caused by the polyethylene which is part of the detector. It is however still clear that the moderation in the container increases the thermal surface current. Table 1 shows the surface current of neutrons below 1 eV and above 1 eV for the different configurations. It can be seen that the detection rate and the rate of neutrons below 1 eV are roughly proportional. Fast neutrons don't have a significant contribution to the detection rate. Thus the container moderation in addition to the polyethylene detector moderation results in an increase of the count rate.

| configuration | Neutron detection rate in He-3 detector [1/s] | Neutrons < 1eV entering He-3 tubes [1/s] | Neutrons > 1eV entering He-3 tubes [1/s] |
|---|---|---|---|
| Bare component | $1.18 \times 10^4$ | $2.15 \times 10^4$ | $5.56 \times 10^4$ |
| 9975 container with Celotex | $1.57 \times 10^4$ | $2.87 \times 10^4$ | $5.06 \times 10^4$ |
| 9975 container with5% borated Celotex | $1.45 \times 10^4$ | $2.65 \times 10^4$ | $5.08 \times 10^4$ |

**Table 1: Neutron detection in the He-3 detector tubes for the different container configurations and neutron rates entering the He-3 tubes**

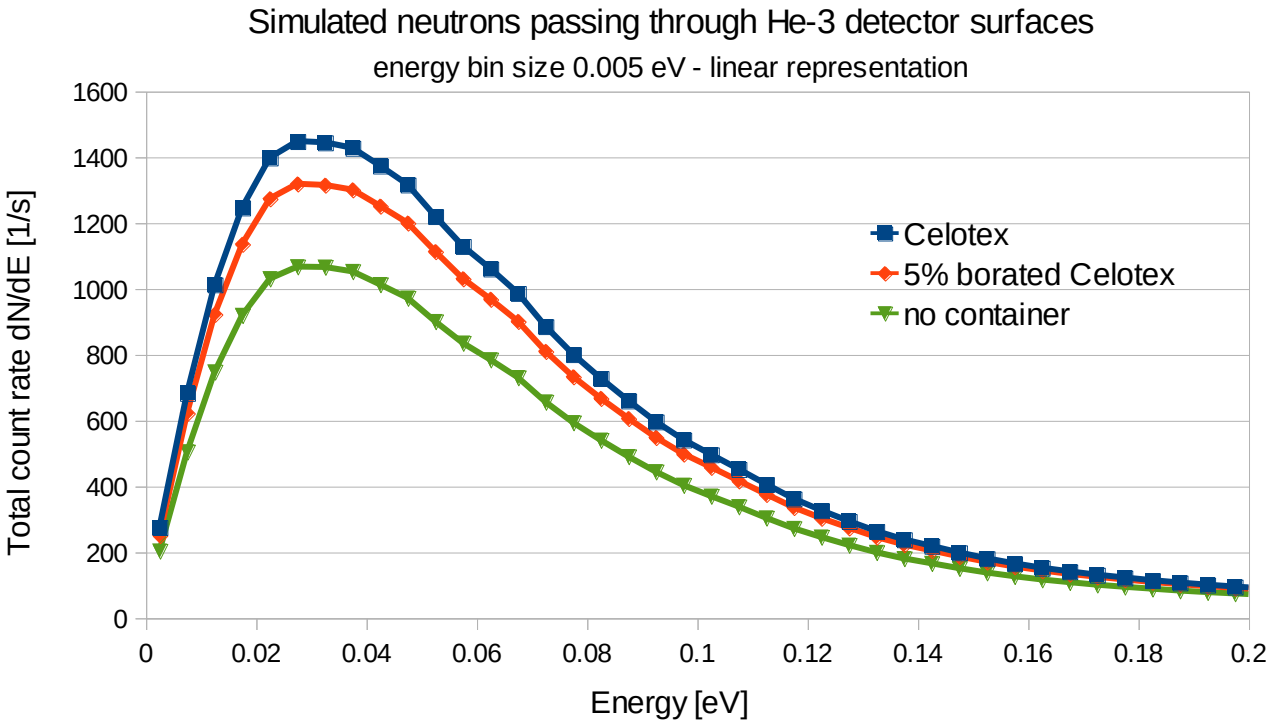


**Figure 3: Energy spectrum of the low-energy neutrons (MCNPX PoliMi simulations) entering the He-3 tubes**

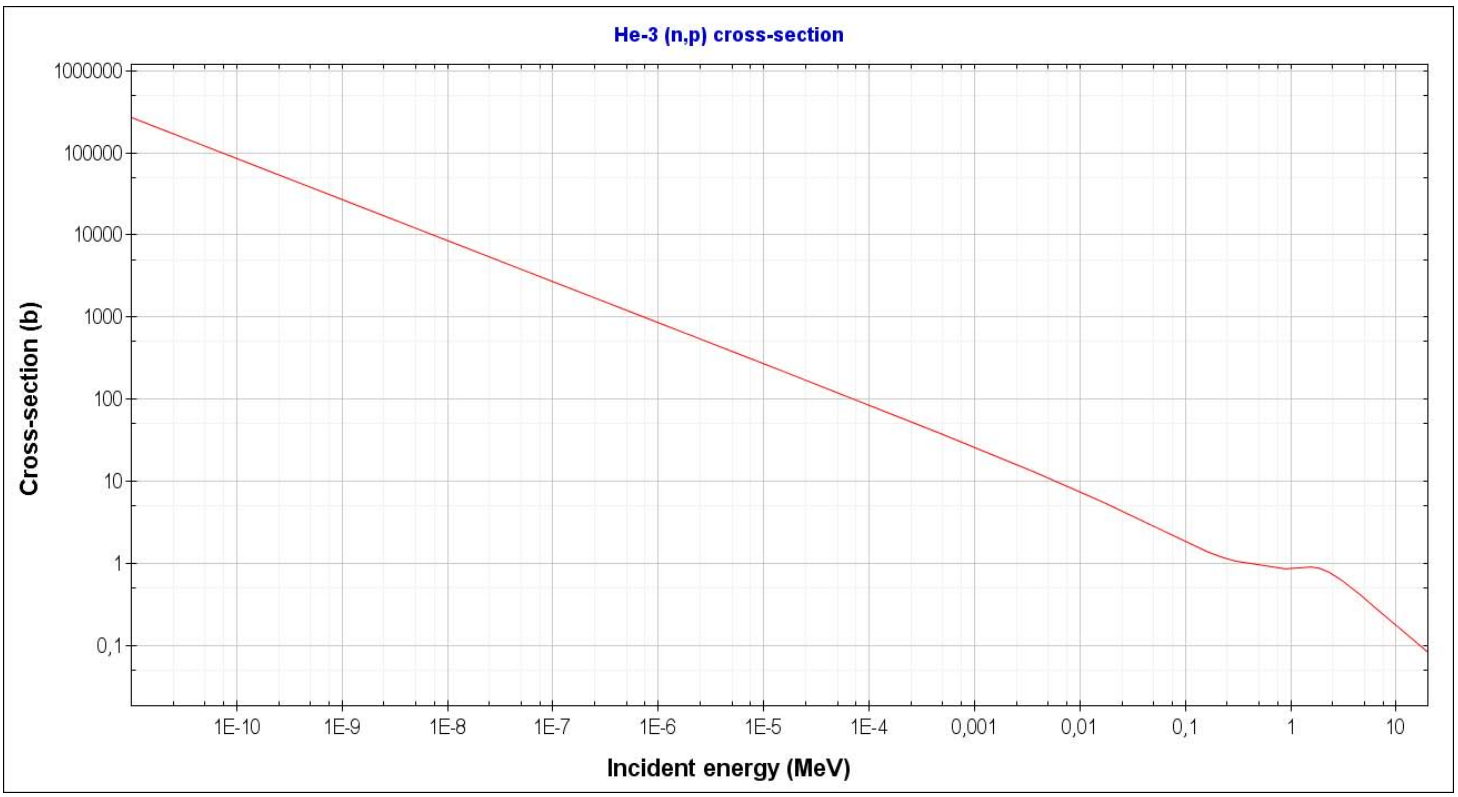


**Figure 4: Cross-section of (n,p) reaction in He-3 from ENDF/B-VII.0**

| | Elastic scattering rate in container [1/s] | Neutron capture rate in container [1/s] |
|---|---|---|
| 9975 container with Celotex | $1.38 \cdot 10^6$ | $7.2 \times 10^3$ |
| 9975 container with 5% borated Celotex | $1.02 \cdot 10^6$ | $2.44 \times 10^4$ (94% by $^{10}B$) |

**Table 2: Elastic scattering and neutron capture events within the containers**

*Neutron capture in container*

Besides elastic collisions, capture reactions occur in the container. Besides capture in Celotex component isotopes, in particular hydrogen, capture also occurs in concrete and stainless steel components of the container. The boron of the borated container has by far the most significant contribution to the overall capture rate. Neutron capture rates for the different container configurations can be compared in Table 2. One can see the large difference between the two configurations, which is due to the boron content which has the highest cross-section for neutron capture.

Looking at the neutron spectrum at the He-3 tube surfaces in Figure 3 and 4, the influence of neutron capture in the container can be seen: The thermal neutron rate of the borated container is 8% lower compared to the non-borated configuration. In the fast neutron energy region, neutron capture does not have any significant influence due to the low reaction cross-section; neutron moderation dominates here.

*Reflection back into fissile material*

Due to scattering, neutrons are scattered back from the container to the plutonium sample where they undergo additional interactions. The energy spectrum of the neutrons that are scattered back and enter the sample is shown in Figure 5. The reason why also the bare component-configuration has a lower but present thermal neutron contribution is that the detector polyethylene scatters back neutrons that can reach the sample.

The effect of back-scattered neutrons is additional induced fission by Pu-239 which creates additional neutrons on the one hand and neutron capture on the other. Looking at Table 3, it can be seen that the effect of induced fission outweighs the effect of neutron capture in terms of net neutron production. This results in a net increase of neutron multiplication. The neutron production rate from the borated Celotex container is rather high, this is under investigation.

| | Neutron production rate from induced fission in fissile material [1/s] | Neutron capture rate in fissile material [1/s] |
|---|---|---|
| Bare component | $1.02 \times 10^5$ | $1.5 \times 10^3$ |
| 9975 container with Celotex | $1.06 \times 10^5$ | $1.7 \times 10^3$ |
| 9975 container with 5% borated Celotex | $1.05 \times 10^5$ | $1.6 \times 10^3$ |

**Table 3: Neutron gain from induced fission and neutron loss from capture in fissile material for different container configuration**

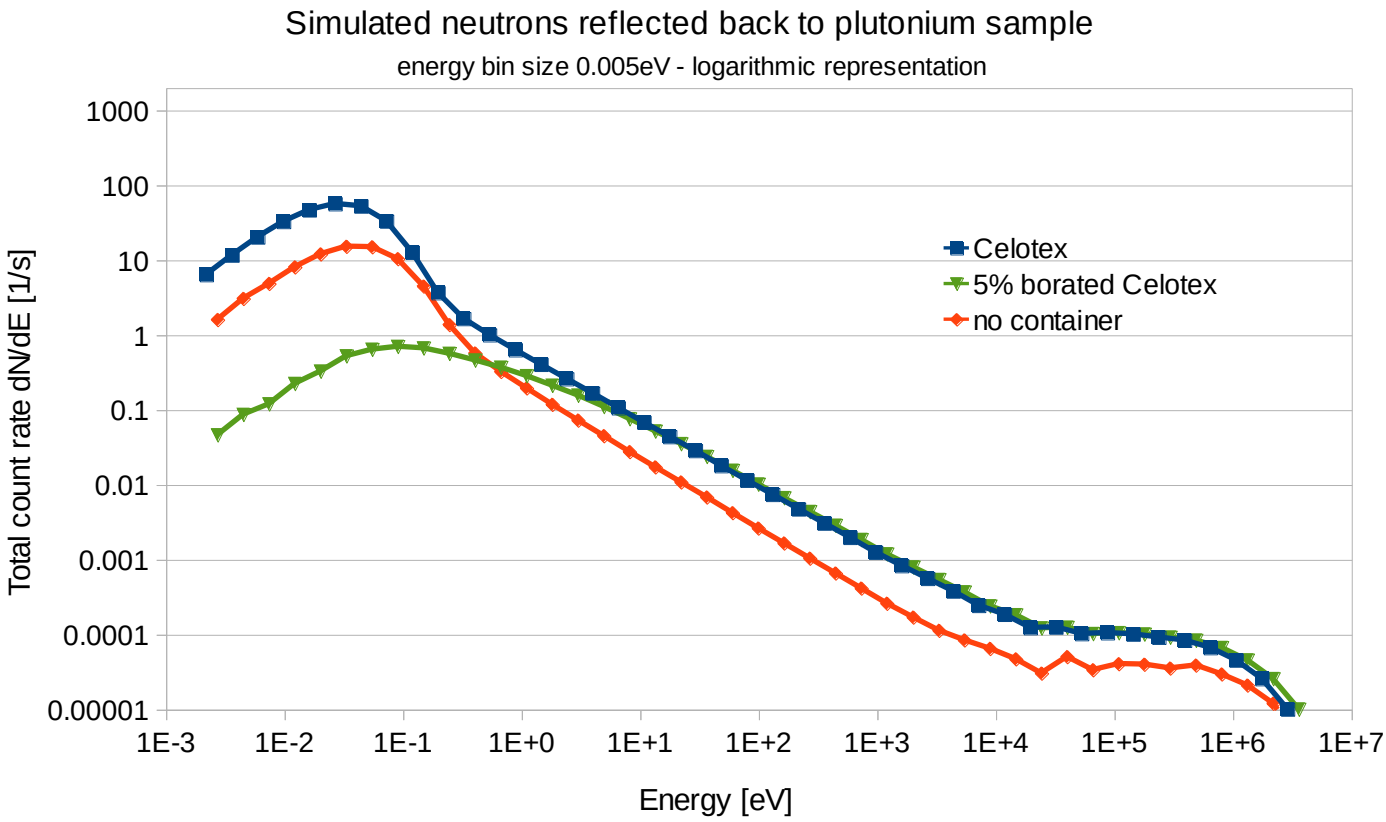


**Figure 5: Logarithmic energy spectrum of all neutron (MCNPX PoliMi simulations) reflected back into fissile material for the different container configurations**

## 3. Conclusion

This study has quantified the neutron processes that occur due the presence of the 9975 container containing a 2 kg weapon-grade plutonium sample. The relevant interactions were found to be neutron moderation in the container (see Fig. 3) that increases the count rate in a He-3 detector, neutron capture in the container that decreases the count rate (see Table 2) as well as reflection back into the fissile sample (see Table 3 and Fig. 5) where the neutrons induce further fission reactions.

These three effects have an influence on neutron multiplicity measurements that could be used for warhead component verification. In future, we shall study the effect of the presence of such a container on the automatic analysis behind an information barrier and to what extent it can decrease the reliability of the information barrier's output.

## 4. Acknowledgements

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# The Analysis and Spectral Assignments of Mixed Actinide Oxide Samples Using Laser-Induced Breakdown Spectroscopy (LIBS)

**James E. Barefield II[1], Elizabeth J. Judge[1], John M. Berg[2], Stephen P. Willson[2], Loan A. Le[3], Leon N. Lopez[1], Leonardo Trujillo[4], David A. Cremers[5], Melissa L. Bostian[5]**

Los Alamos National Laboratory
PO Box 1663, MS J565
Los Alamos, NM 87545
[1]Chemical Diagnostics and Engineering
[2]Manufacturing Engineering and Technology
[3]Physical Chemistry and Applied Spectroscopy
[4]Actinide Process Chemistry
[5]Applied Research Associates, Albuquerque, NM
E-mail: jbarefield@lanl.gov

***Abstract:***
In this paper, we report the identification and assignments of complex atomic emission spectra of mixed actinide oxides using laser-induced plasma spectroscopy or laser-induced breakdown spectroscopy (LIBS). Results of LIBS measurements on samples of: (1) $UO_2$ / $ThO_2$ in a stearic acid binder, (2) $UO_2$ / $PuO_2$ / $AmO_2$ / $NpO_2$ in simulated fuel pellets (or mixed actinide oxide sample), (3) $UO_2$ / $PuO_2$, (4) isotopic samples of uranium oxide, (5) preliminary single shot measurements of depleted uranium, and (6) a vitrified glass bead sample are reported and discussed. Over 800 spectral lines (transitions) have been identified and assigned for the pressed and fuel pellet samples thus far. The identification and assignments of spectral emission transitions for Th, U, Pu, and Am are consistent with wavelength data from the literature. However, only a few transitions have been assigned with a high degree of confidence for Np when compared to available atomic emission data from the literature. Preliminary isotopic measurements on samples of enriched uranium ($^{235}U$) between 1 and 97 % are also reported in this paper. We also indicate where atomic emission transitions for curium (Cm) would most likely appear in the displayed spectral regions shown. This work clearly indicates that a LIBS system with a resolving power of approximately 20,000 is adequate for analyzing complex mixtures of actinide elements within the same sample.

**Keywords: Forensics; Spectral Assignments; Mixed Actinide Oxide Samples; LIBS**

## 1. Introduction

The interest in laser-induced breakdown spectroscopy (LIBS) analysis of actinide-containing samples stems from the need for a rapid, accurate, and sensitive field deployable instruments for global security, nuclear forensics, environmental sampling, site characterization, clean-up validation, and hazardous waste characterization. Immediately following a detonation event, breach of containment, or use involving nuclear material, emergency response personnel and inspectors require tools and instruments to determine the nature of the explosive as well as any indication as to where the device or material originated. A field portable LIBS system could be used to determine the degree of enrichment of the nuclear material as well as trace elements present that would help to identify or, at a minimum, reduce the number of possible sources of the material. Traditional analytical methods require sample collection, packaging, and shipment to an accredited laboratory before analysis can take place, which could take

days, weeks, and sometimes months to complete. Therefore, there is great interest in developing a rapid, in-field technique capable of determining the type of material present and possibly the origin as well.

Results from several investigations using LIBS to analyze samples containing single actinides in different matrices have been reported [1-4]. LIBS has also been used to analyze samples of enriched uranium and plutonium oxide [5-7] . As a step toward using LIBS to analyze samples in the field from post-detonation events, fuel cycle monitoring, and in general actinide analysis, the analysis of mixed actinide fuel pellets allows for the identification and assignment of actinide emission lines under known conditions.

LIBS analysis of actinide elements involves a complicated and complex set of excited state quantum physics and photo-dynamics between the high densities of excited states (both within and between the atoms in the plasma) that is the origin of the observed atomic emission spectra. Furthermore, the observed spectra are the results of emission from excited states to the ground state and emission between excited states. In addition, the elements in the chemical environment or matrix also contribute to the excited state quantum physics and photo-dynamics and therefore add yet another degree of complexity to the observed spectra. Finally, in many analysis cases only small quantities of sample are available. For those cases, a rapid analysis method is also desirable. Preliminary results from LIBS analysis on microgram quantities of samples and single particle analysis are also reported and will be discussed.

## 2. Experimental

For most of the results reported in this paper, a LIBS system consisting of an LLA ESA 3000 or 4000 Echelle spectrometer (LLA Instruments GmbH, Berlin, Germany), with a reported resolving power of 20, 000, was used to collect the mixed actinide data. The excitation source was an Nd:YAG laser (Quantel USA), with a maximum output of 100 mJ per pulse operating at a repetition rate of 20 Hz (shots per second) and a pulse width of 7 ns. The delay time was varied between 1 and 4 microseconds. Typically, we used between 15 and 60 mJ per pulse of excitation energy for the mixed oxide samples. For the uranium isotopic measurements, we used approximately 10 mJ of excitation energy from a diode pumped Nd:YAG laser (JP Innovations model DP 120; 1064 nm, 14 mJ / pulse, 10 Hz, 7-9 ns pulse width). For the single particle preliminary experiments, we used approximately 30 mJ of excitation energy from a Nd:YAG laser system (Quantel USA) operated at 10 Hz or in some cases single shot mode. The single shot spectra were collected with an EMU 65 Echelle spectrometer (Catalina Scientific, Tucson Arizona). This spectrometer allows a range of resolving powers ranging up to 60, 000 depending upon the installed dispersion cassette. The dispersion cassette that we used had a resolving power of 53, 000. Although, this spectrometer can be used to measure isotopic samples of uranium, for the preliminary uranium isotopic measurements shown here, we used a high resolution Echelle spectrometer (LTB LaserTechnik, Berlin Germany) with a reported resolving power of 75, 000.

## 3. Results and Discussion

We have collected LIBS spectral data for the following samples: (1) $UO_2$ / $ThO_2$ in a stearic acid binder, (2) $UO_2$ / $PuO_2$ / $AmO_2$ / $NpO_2$ in simulated fuel pellets (or mixed actinide oxide sample), (3) $UO_2$ / $PuO_2$, (4) isotopic samples of uranium oxide, (5) preliminary single shot measurements of depleted uranium, and (6) a vitrified glass bead sample.

### 3.1 LIBS spectra and assignments of low radioactive depleted uranium and thorium oxide pressed pellet samples.

LIBS was used to analyze samples of $ThO_2$, $DUO_2$, and a 50 / 50 $ThO_2$ / $DUO_2$ mixture to determine if emission spectra collected from thin layers of powder on carbon adhesive discs were comparable to those obtained from pressed pellets. Emission lines were assigned for all samples by comparison to high

resolution atomic emission data from the literature [8-10]. Thorium oxide [11] and depleted uranium oxide [12] pellets were previously analyzed using LIBS. Over 3000 emission lines were assigned between 200 nm and 780 nm for thorium and uranium. Here only small spectral regions are shown for comparison between the emission spectrum from powder on carbon adhesive discs and pressed pellets. A 50/50 (%wt) mixture of $ThO_2$ and $DUO_2$ was analyzed on carbon adhesive discs and in pellet form. Figure 1a and 1b show the emission spectra for the 50/50 mixture of $ThO_2$ and $DUO_2$ pellet and powder between 384 and 389 nm, respectively. For a complete spectral assignment of the actinide pellets (U, Th) from 200 – 780 nm, see references [11-12].

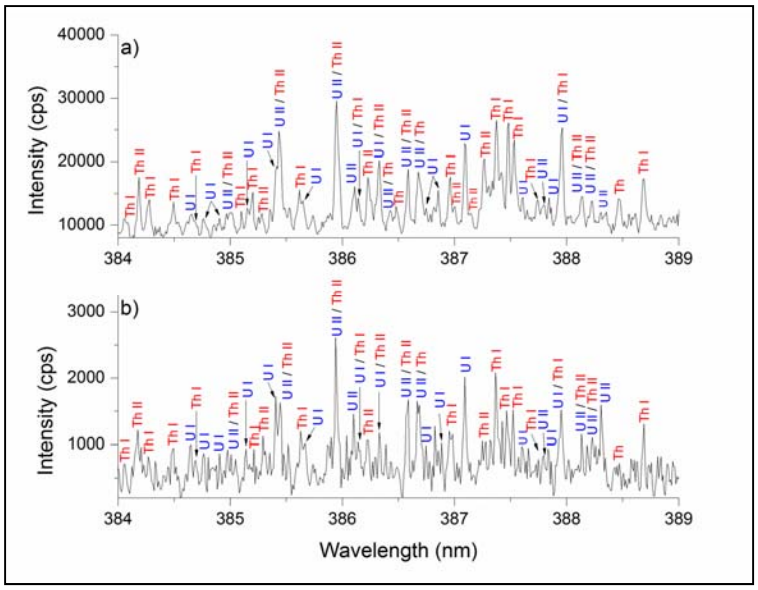


**Figure 1**. LIBS Spectra of 50/50 $ThO_2/DUO_2$ (a) pellet and (b) powder on a carbon adhesive disc are shown between 384 and 389 nm.

In the 50/50 mixture of $ThO_2$ and $DUO_2$ spectra (Figure 1), uranium and thorium spectral features were observed, identified, assigned, and can be found in Table 1. However, the relative intensities of the uranium and thorium spectral features change when the two elements are present in the same sample compared to the spectral intensities when analyzed alone [11-12]. Since both actinides have very complex and dense electronic excited state structures, the actinides compete for the excitation energy as well as undergo interactions and reactions with each other within the plasma. A more complete discussion can be found in reference [13].

| Thorium Oxide and Uranium Oxide | | | | | | | |
|---|---|---|---|---|---|---|---|
| | ***Literature (nm)*** | | | ***Literature (nm)*** | | | ***Literature (nm)*** |
| Th I | 384.0800 | | U II | 385.957 | | Th II | 387.14563 |
| Th II | 384.196 | | Th II | 385.984 | | Th II | 387.2722 |
| Th I | 384.2897 | | U II | 386.117 | | Th I | 387.3822 |
| Th I | 384.50924 | | Th I | 386.15023 | | U II | 387.40387 |
| Th II | 384.55369 | | U I | 386.17581 | | Th I | 387.42438 |
| U I | 384.6553 | | Th II | 386.23972 | | U I | 387.4458 |
| Th I | 384.6887 | | U I | 386.30935 | | Th I | 387.4862 |
| U I | 384.7833 | | Th II | 386.3405 | | Th I | 387.5374 |
| U II | 384.86042 | | Th* | 386.43744 | | U I | 387.6133 |
| U I | 384.92146 | | U II | 386.4467 | | Th I | 387.75564 |
| U II | 384.9847 | | Th* | 386.48381 | | U II | 387.80847 |
| Th II | 385.0134 | | U II | 386.592 | | U I | 387.85738 |
| Th I | 385.10746 | | Th II | 386.60683 | | Th I | 387.92688 |
| U I | 385.17268 | | U II | 386.680 | | Th I | 387.9644 |
| Th I | 385.2135 | | Th* | 386.68741 | | U I | 387.9711 |
| Th II | 385.2959 | | U I | 386.71716 | | Th I | 388.01948 |
| U I | 385.422 | | U I | 386.75047 | | U II | 388.14546 |
| Th II | 385.4511 | | Th II | 386.78505 | | Th II | 388.1498 |
| U II | 385.464 | | U I | 386.85164 | | Th II | 388.2143 |
| U I | 385.54282 | | U I | 386.86969 | | U II | 388.23556 |
| Th I | 385.63544 | | Th I | 386.9663 | | U II | 388.328 |
| U I | 385.67413 | | Th II | 386.9971 | | Th* | 388.48225 |
| U II | 385.90082 | | U I | 387.10353 | | Th I | 388.6915 |

**Table 1.** Spectral assignment of the emission lines observed in 50/50 $ThO_2$/$DUO_2$ pellet and powder samples based on atomic emission data found in the literature are listed.

## 3.2 LIBS spectra and assignments of $UO_2$ / $PuO_2$ / $AmO_2$ / $NpO_2$ fuel pellet samples

In the laser-generated plasma, when many atomic species are present, many interactions and reactions can occur, thereby reducing or increasing the relative intensity of an emission line. Therefore, the relative intensity of a single spectral feature cannot be used for quantification determinations without the chemical environment and excited state structure of the species present in the plasma being carefully considered. Figure 2a-b shows several expanded regions of the LIBS spectrum of U/Pu/Am/Np simulated fuel pellet. When small regions of the spectrum for the U/Pu/Am/Np fuel pellet are selected and expanded, emission lines can be identified and assigned to the actinide elements via comparison with high-resolution atomic emission data from the literature. A more complete analysis and discussion can be found in references [14, 15].

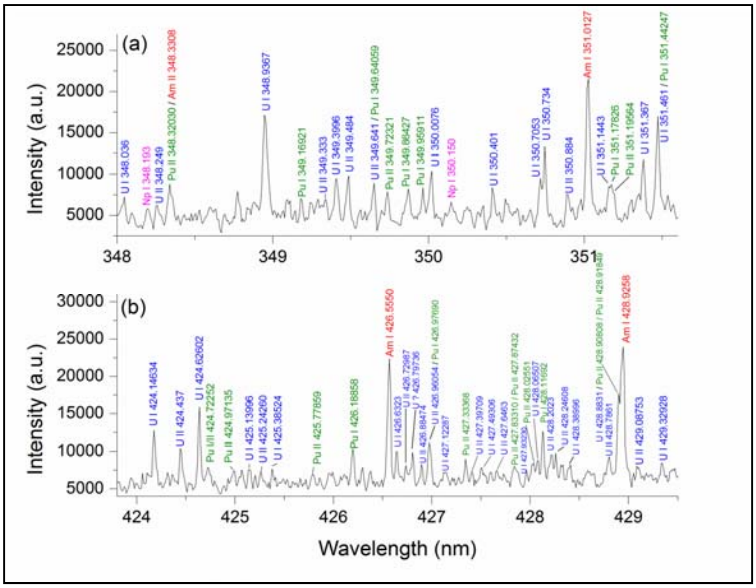


**Figure 2**. Selected expanded regions of a LIBS spectrum of a U/Pu/Am/Np mixed oxide sample between (a) 348.0-351.6 nm and (b) 423.8 – 429.5 nm.

Table 2 consists of the identified emission lines and assignments for the actinide elements in the mixed actinide oxide sample. For a more complete set of emission lines and assignments, see Judge et al.[15]. In that report, we identified and assigned 800 emission lines for the actinide atoms (U, Pu, Am, and Np) in the mixed actinide oxide sample. The spectral regions shown here were chosen based on the probability that all of the actinides (U, Pu, Am, and Np) could be observed within the same narrow spectral window. There were many more uranium and plutonium emission lines in comparison with americium and neptunium. This difference is most likely due to the relative abundance of uranium and plutonium being much greater than that of americium and neptunium in the fuel pellet sample (approximate composition: 65% $UO_2$ / 30% $PuO_2$ / 3% $AmO_2$ / 2% $NpO_2$). More than 30 americium emission lines were assigned compared with four for neptunium. There may be several reasons for the small number of emission lines observed for neptunium. First, the chemical compositions of the fuel pellets used in these experiments are approximate weight percents. Therefore, the actual composition of neptunium in the sample may be less than 2%. Secondly, the excited state levels of neptunium are low and might be undergoing more interactions and reactions with the other actinides present in the laser generated plasma, such as energy transfer, collisional deactivation, or excited atom–excited atom interactions. There are also many other factors that must be considered when trying to unravel the plasma dynamics and resulting emission spectra but are beyond the scope of this paper. However, experiments are being developed to better understand the role of each actinide in the plasma and how the emission spectra can aid in that understanding.

| U/Pu/Am/Np Mixed Oxide Sample | | | | | | | |
|---|---|---|---|---|---|---|---|
| | ***Literature (nm)*** | | | ***Literature (nm)*** | | | ***Literature (nm)*** |
| U I | 348.036 | | U I | 351.1443 | | Pu I | 426.9769 |
| Np I | 348.193 | | Pu I | 351.17826 | | U I | 427.12287 |
| U II | 348.249 | | Pu II | 351.19564 | | Pu II | 427.33368 |
| Pu II | 348.3203 | | U I | 351.367 | | U II | 427.39709 |
| Am II | 348.3308 | | Pu I | 351.44247 | | U I | 427.49306 |
| U I | 348.9367 | | U I | 351.461 | | U II | 427.6463 |
| Pu I | 349.16921 | | U I | 424.14634 | | Pu II | 427.8331 |
| U II | 349.333 | | U II | 424.437 | | Pu II | 427.87432 |
| U I | 349.3996 | | U I | 424.62602 | | U I | 427.9323 |
| U II | 349.484 | | Pu I/II | 424.72252 | | Pu II | 428.02551 |
| Pu I | 349.64059 | | Pu II | 424.97135 | | U I | 428.06507 |
| U II | 349.641 | | U I | 425.13996 | | Pu I | 428.11692 |
| Pu II | 349.72321 | | U II | 425.2426 | | U II | 428.2023 |
| Pu I | 349.86427 | | U I | 425.38524 | | U II | 428.24608 |
| Pu I | 349.95911 | | Pu II | 425.77859 | | U I | 428.38996 |
| U I | 350.0076 | | Pu I | 426.18858 | | U II | 428.7861 |
| Np I | 350.15 | | Am I | 426.555 | | U I | 428.8831 |
| U I | 350.401 | | U I | 426.6323 | | Pu II | 428.90808 |
| U I | 350.7053 | | U II | 426.72987 | | Pu II | 428.91849 |
| U I | 350.734 | | U | 426.79736 | | Am I | 428.9258 |
| U II | 350.884 | | U II | 426.88474 | | U II | 429.08753 |
| Am I | 351.0127 | | U II | 426.96054 | | U I | 429.32928 |

**Table 2.** Spectral assignment of the emission lines observed in U/Pu/Am/Np mixed oxide sample based on atomic emission data found in the literature are listed.

### 3.3 LIBS spectra and assignments of $UO_2$ / $PuO_2$ simulated fuel pellet samples

A $UO_2/PuO_2$ simulated fuel pellet was also measured using LIBS and a represented set of data is shown in Figure 3. Table 3 contains an abbreviated list of the spectral features assigned in the wavelength regions shown in Fig. 3a-b (348.0 - 351.6 nm and 423.8 - 429.5 nm). It is easily seen that the americium and neptunium lines are missing in the $UO_2/PuO_2$ simulated fuel pellet spectra, as expected. However, there are a few new uranium and plutonium features that appear in the $UO_2/PuO_2$ fuel pellet data that were not present (or in lower signal intensity) in the mixed actinide oxide fuel pellet data: 425.47616 nm Pu(II), 468.21210 nm U(I), 468.58123 nm U(I), and 468.69226 nm U(I). These features could possibly be explained by the difference in plasma dynamics that occur when the chemical environment of the sample being analyzed is changed. It is very important to investigate the difference in spectral features due to the chemical environment when considering the use of a LIBS system. Different spectral features may be observed in the presence of different elements. Further understanding of these dynamics is required to

produce a robust, accurate, and field-portable instrument capable of analyzing complex actinide samples based on changes in the chemical environment.

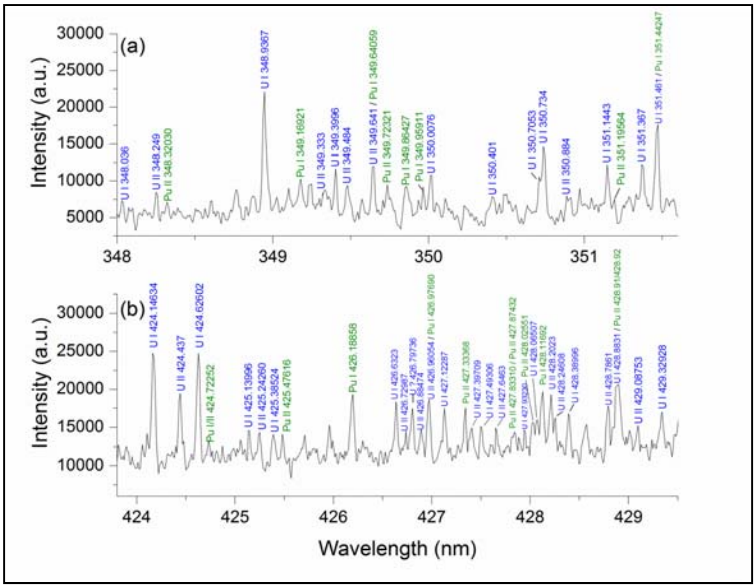


**Figure 3**. Selected expanded regions of a LIBS spectrum of $UO_2/PuO_2$ simulated fuel pellet sample between (a) 348.0 – 351.6 nm and (b) 423.8-429.5 nm

| $UO_2$ / $PuO_2$ Simulated Fuel Pellet | | | | | | | |
|---|---|---|---|---|---|---|---|
| | ***Literature (nm)*** | | | ***Literature (nm)*** | | | ***Literature (nm)*** |
| U I | 348.036 | | U I | 351.367 | | U II | 427.3971 |
| U II | 348.249 | | Pu I | 351.4425 | | U I | 427.4931 |
| Pu II | 348.3203 | | U I | 351.461 | | U II | 427.6463 |
| U I | 348.9367 | | U I | 424.1463 | | Pu II | 427.8331 |
| Pu I | 349.16921 | | U II | 424.437 | | Pu II | 427.8743 |
| U II | 349.333 | | U I | 424.626 | | U I | 427.9323 |
| U I | 349.3996 | | Pu I/II | 424.7225 | | Pu II | 428.0255 |
| U II | 349.484 | | U I | 425.14 | | U I | 428.0651 |
| Pu I | 349.64059 | | U II | 425.2426 | | Pu I | 428.1169 |
| U II | 349.641 | | U I | 425.3852 | | U II | 428.2023 |
| Pu II | 349.72321 | | Pu II | 425.4762 | | U II | 428.2461 |
| Pu I | 349.86427 | | Pu I | 426.1886 | | U I | 428.39 |
| Pu I | 349.95911 | | U I | 426.6323 | | U II | 428.7861 |
| U I | 350.0076 | | U II | 426.7299 | | U I | 428.8831 |
| U I | 350.401 | | U | 426.7974 | | Pu II | 428.9081 |
| U I | 350.7053 | | U II | 426.8847 | | Pu II | 428.9185 |
| U I | 350.734 | | U II | 426.9605 | | U II | 429.0875 |
| U II | 350.884 | | Pu I | 426.9769 | | U I | 429.3293 |
| U I | 351.1443 | | U I | 427.1229 | | | |
| Pu II | 351.19564 | | Pu II | 427.3337 | | | |

**Table 3.** Spectral assignment of the emission lines observed in $UO_2$/$PuO_2$ simulated fuel pellet sample based on atomic emission data found in the literature are listed.

One of the intended uses of this type of LIBS system is to measure elemental ratios in spent fuel samples that contain mixtures of actinide elements including uranium, plutonium, curium, and fission products. Therefore, the location of the atomic emission transitions for curium are of interest, for example within the same spectral regions of the LIBS spectra data for samples of U / Pu / Am / Np and U / Pu. To address this issue, we considered the atomic emission spectral data for curium from the literature [16, 17]. From the curium spectral emission data tables of Worden [17], we identified 12 possible curium transitions that would be located within the expanded regions of the spectra shown in Figures 2 and 3. The transitions along with ionic state labels (neutral or ionic) are listed in Table 4. Although the data of Worden was recorded using hollow cathode lamps at low pressure and therefore involve a different set of photo-dynamics and quantum physics compared to the laser-induced plasma used in LIBS measurements, the reference provides a list of possible emission lines that may be observed. The observation of the curium atomic emission transitions using LIBS also depends upon the Einstein coefficient for emission (or probability of emission) from the excited states of curium atoms in the plasma, concentration of curium in the sample, and sufficient laser energy density to excite transitions in the curium atom in the plasma. We are, however, convinced that if curium is present in a mixed actinide sample at a sufficient concentration, then a LIBS system with a resolution of 20,000 would be adequate in determining the presence and the concentration of curium in a complex sample.

| Curium Emission Lines | |
|---|---|
| | ***Literature (nm)*** |
| Cm II | 347.31 |
| Cm I/II | 348.24 |
| Cm I | 348.78 |
| Cm II | 349.88 |
| Cm I | 350.39 |
| Cm I | 351.03 |
| Cm I | 451.78 |
| Cm I | 452.66 |
| Cm I | 453.82 |
| Cm I | 456.03 |
| Cm I | 456.39 |
| Cm I | 469.5 |

**Table 4.** Curium emission lines that may appear in the spectral regions discussed in this article and reference [14].

### 3.4 LIBS spectra and analysis of enriched uranium oxide samples

We have also used LIBS to analyze isotopic uranium oxide powder samples on carbon adhesive discs ranging in enrichments from 1 to 97 percent uranium (235). A representative set of LIBS data is shown in Figure 4 for 35 and 50 percent enrichments. The data, as shown, includes significant background noise. There are several reasons for this. First, the system used was set up to collect data from 60 laser shots. However, since the powder samples were fixed to carbon adhesive discs, only about the first 5-10 laser shots contained spectral data. The remaining 50 laser shots added noise to the 60 shot averaged spectral data collected for a given enrichment. Secondly, the powder samples were not distributed uniformly over the surface of the carbon adhesive disc. Therefore, each laser shot did not interrogate the same amount of enriched uranium sample. Finally, the estimated amount of sample analyzed per enrichment was approximately 1 microgram. The samples and compositions are listed in Table 5. A preliminary set of analyses were performed and the calibration curve is shown in Figure 5. In light of the difficulties mention above, the preliminary data is encouraging indicating that a reasonable calibration curve ($R^2$ = 0.9931) can be obtained using microgram quantities of enriched uranium powder adhered to a carbon adhesive disc. We are in the process of modifying the data acquisition and analysis code to collect data from fewer laser shots and trying to achieve a more uniform coverage of the samples on the carbon discs. The results of this effort should be spectra with a higher signal-to-noise ratio.

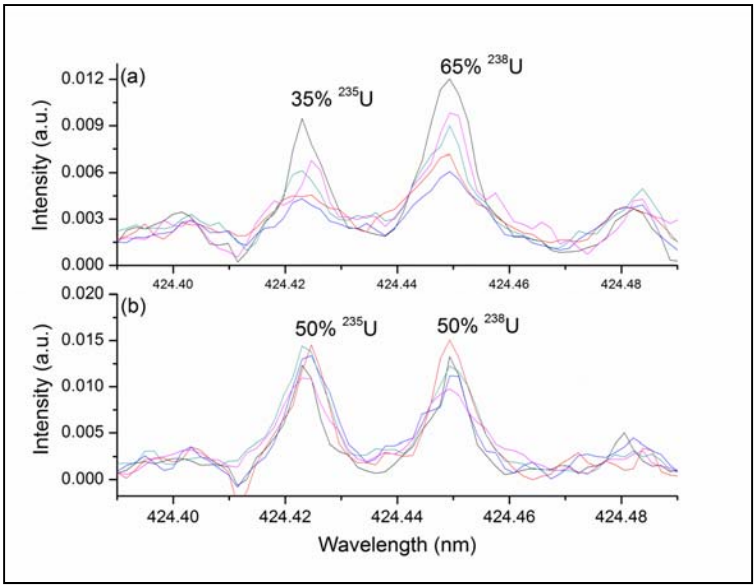


**Figure 4**. LIBS spectra of enriched uranium (a) 35% and (b) 50%, showing the isotopic shift of the $^{235}U$ and $^{238}U$ emission lines at 424.412 nm and 424.437 nm, respectively.

| Sample ID | $^{235}U$ (wt %) | $^{238}U$ (wt %) |
|---|---|---|
| U-1 | 1.0037 | 98.984 |
| U-1.5 | 1.5323 | 98.443 |
| U-2 | 2.038 | 97.933 |
| U-3 | 3.046 | 96.915 |
| U-5 | 5.01 | 94.915 |
| U-10 | 10.19 | 89.704 |
| U-15 | 15.307 | 84.528 |
| U-20 | 20.013 | 79.654 |
| U-35 | 35.19 | 64.393 |
| U-50 | 49.566 | 49.711 |
| U-75 | 75.357 | 23.801 |
| U-80 | 80.279 | 18.82 |
| U-85 | 85.137 | 13.848 |
| U-90 | 90.196 | 8.693 |
| U-93 | 93.336 | 5.38 |
| U-97 | 97.663 | 0.5229 |

**Table 5.** Sample composition of enriched uranium powder.

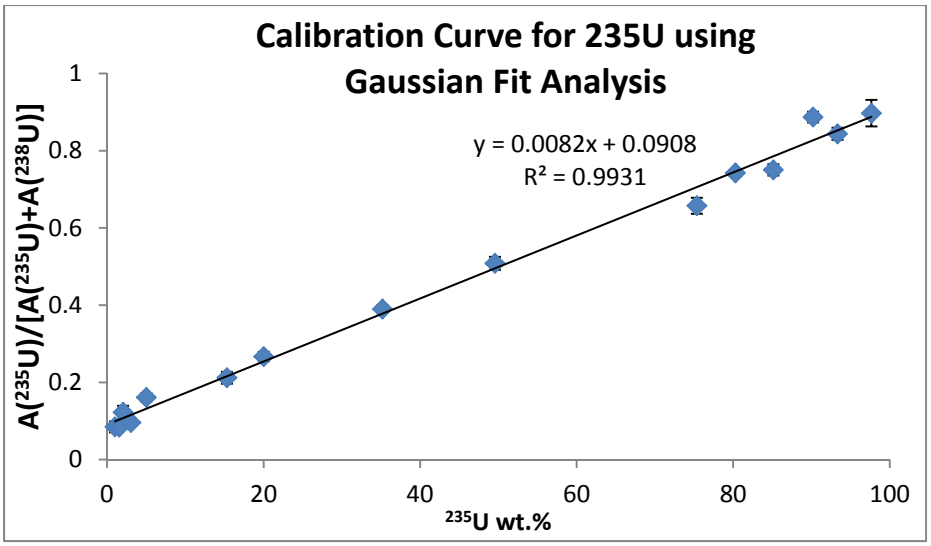


**Figure 5.** Calibration curve for enriched uranium ($^{235}$U) using a Gaussian fit analysis.

### 3.5 LIBS spectra and assignments of a single shot depleted uranium sample

We used a LIBS system containing a high resolution spectrometer form Catalina Scientific (EMU 65) to collect single shot spectra of samples of depleted uranium. Representative spectra at two different resolving powers (53,050 and 14,150) are shown in Figure 6. The spectral transition shown in the middle of the spectra located at 424.437 nm has the largest isotope shift for uranium enriched in $^{235}$U relative to $^{238}$U in this region. The isotope shift is 25 pm (picometers). Therefore, we believe that we can use this system to analyze enriched samples of uranium at the single shot level. However, environmental sampling where the enriched samples will be contained in challenging matrices like soil, metal surfaces, etc., will still need to be evaluated before a field deployable robust system can be used in a field setting.

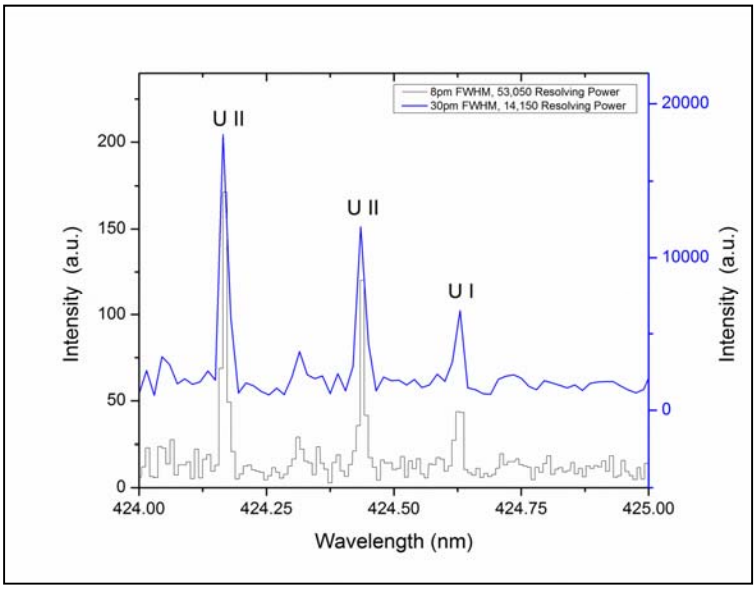


**Figure 6**. LIBS spectrum of depleted uranium using a Catalina Scientific EMU-65 spectrograph with 53,050 and 14,150 resolving power.

### 3.6 LIBS spectra and assignments of a single vitrified bead sample or particle

Finally**,** we used the LIBS system discussed in *Section 3.5* above to analyze single particles of a vitrified sample that could possibly contain actinide elements. The spatial resolution of this system is approximately 25 microns. Spectra of a single particle of a vitrified glass bead are shown in Figure 7 along with assignments listed in Table 6. Again, we believe that we can use this system to analyze single particle samples that could possibly contain actinide elements down to sizes on the order of 25 microns. This could be a useful tool for an inspector to analyze particles on swipe material.

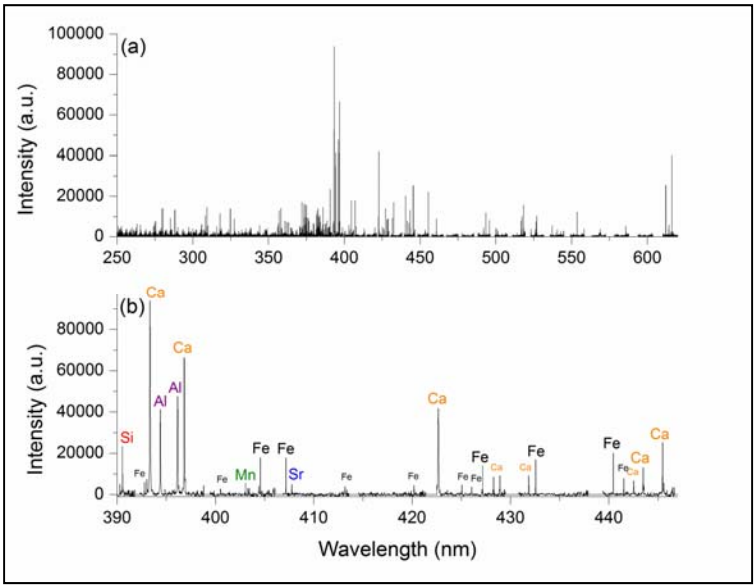


**Figure 7**. LIBS spectrum of a vitrified pellet, (a) full spectrum and (b) zoomed view between 390 - 447 nm.

| Vitrified Glass Bead | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | ***Literature (nm)*** | | | ***Literature (nm)*** | | | ***Literature (nm)*** | |
| Fe I | 390.295 | | Fe I | 404.581 | | Fe I | 428.240 | |
| Si I | 390.552 | | Fe I | 407.174 | | Ca I | 428.302 | |
| Fe I | 392.792 | | Sr II | 407.771 | | Ca I | 428.936 | |
| Fe I | 393.03 | | Fe I | 413.206 | | Ca I | 431.865 | |
| Ca II | 393.366 | | Fe I | 420.203 | | Fe I | 432.576 | |
| Al I | 394.401 | | Ca I | 422.673 | | Fe I | 440.475 | |
| Al I | 396.152 | | Fe I | 423.594 | | Fe I | 441.512 | |
| Ca II | 396.847 | | Fe I | 425.012 | | Ca I | 442.544 | |
| Fe I | 400.524 | | Fe I | 425.079 | | Ca I | 443.497 | |
| Mn I | 403.076 | | Fe I | 426.047 | | Ca I | 443.57 | |
| Mn I | 403.307 | | Fe I | 427.115 | | Ca I | 445.478 | |
| Mn I | 403.449 | | Fe I | 427.176 | | Ca I | 445.589 | |

**Table 6.** Spectral assignment of the emission lines observed in the vitrified glass bead based on atomic emission data found in the literature are listed.

## 4. Conclusions

In this work we have shown that a medium resolution system is adequate for measuring complex mixtures of actinides in the same sample. The more challenging task is to determine the optimal set of experimental parameters that will yield the best LIBS spectra for the mixed actinide oxide samples and to optimize the precision and accuracy of the measurements. For example, we have identified and assigned several strong and isolated emission lines in the LIBS spectra of the actinide elements. We need to determine which emission lines give the best analytical performance. In a broader sense, due to the complex set of quantum physics and photo-dynamics that is the origin of the observed LIBS spectra of the actinide elements, the spectra of the mixtures are not the numerical sum of the individual single-element spectra. One of the intended uses of a medium-resolution LIBS system is in the analysis of nuclear and spent fuel to determine elemental ratios. Based upon our LIBS measurements using a medium-resolution LIBS system both in a low and high radioactivity environment, analysis of nuclear fuel and spent nuclear fuel can be performed. A low resolution system would not be useful in resolving the dense spectral features observed due to the complex excited state levels and associated plasma dynamics of the actinides Furthermore, analysis of important ratios such as, for example, U/Cm, Pu/Cm, U/Pu, and Pu/U can be accomplished using a medium-resolution LIBS system depending upon the minimum concentration of the elements of interest.

## 5. Acknowledgements

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# Expanding the Capabilities of Neutron Multiplicity Measurements: Conclusions from a Four Year Project

**Braden Goddard[1], William Charlton[1], Martyn Swinhoe[2], Paolo Peerani[3]**

[1]Nuclear Security Science and Policy Institute, Texas A&M University
College Station, Texas, USA
E-mail: goddard.braden@gmail.com, wcharlton@tamu.edu

[2]Los Alamos National Laboratory, NEN-1,
Los Alamos, New Mexico, USA
E-mail: swinhoe@lanl.gov

[3]European Commission, EC-JRC-IPSC
Ispra, Italy
E-mail: paolo.peerani@jrc.ec.europa.eu

***Abstract:***

*Neutron multiplicity measurement techniques are a powerful nondestructive assay tool for the quantification of plutonium or uranium. One of the main limitations of this technique is the ability to quantify the content of a sample which contains more than one actinide. The ability of inspection agencies and facility operators to measure powders containing several actinides is increasingly necessary as new reprocessing techniques and fuel forms are being developed. These powders are difficult to measure with current techniques because neutrons emitted from induced and spontaneous fission of different nuclides are very similar. Over the past four years at the Nuclear Security Science and Policy Institute (NSSPI), at Texas A&M University, a neutron multiplicity technique based on first principle methods was developed to measure these powders by exploiting isotope-specific nuclear properties, such as the energy-dependent fission cross sections and the neutron induced fission multiplicity. This technique was tested by three measurement campaigns using the Active Well Coincidence Counter (AWCC) and Epithermal Neutron Multiplicity Counter (ENMC) with various (α,n) sources and measured materials. To complement these measurements, extensive Monte Carlo N Particle eXtended (MCNPX) simulations were performed for each measured sample, as well as samples which were not available to measure. Four potential applications of this first principal technique have been identified: measurements of U, Np, Pu, and Am materials, mixed oxide (MOX) materials, and uranium materials, as well as weapons verification in arms control agreements. This technique still has several challenges which need to be overcome, the largest of these being the ability to produce results with acceptably small uncertainties.*

**Keywords:** Neutron Multiplicity Counting; Epithermal Neutron Multiplicity Counter (ENMC); Monte Carlo N Particle Extended (MCNPX); First Principle Methods; Nondestructive Assay

## 1. Introduction

Neutron multiplicity counting is a well developed technique which is commonly used to quantify nuclear materials. The three primary materials that are measured with this technique are: plutonium in plutonium samples, plutonium in mixed oxide (MOX) samples, and $^{235}U$ in uranium samples. Although these materials currently represent the majority of unirradiated nuclear materials which are safeguarded, there is a growing need to develop techniques to measure more complex materials. As new reprocessing methods [1][2] and fuel forms [3-8] are developed, the ability to perform neutron multiplicity measurements on materials which contain other actinides, including neptunium and americium, are becoming increasing important. In addition to measuring more complex materials,

there is an additional need to redevelop the equations and theory of active neutron multiplicity counting to increase the possible applications for this technique. By moving from an active measurement technique which requires known standards of similar composition and geometry to a method which is based on first principles, a wider range of materials can be measured.

## 2. Theory

Active neutron coincidence counting is currently performed by interrogating the measurement sample with neutrons from an (α,n) source, such as americium-lithium (AmLi). Because neutrons created from (α,n) reactions are created individually they have a minimal impact on the doubles count rate. Instead of using the detector's correlated count rates to solve for sample related variables, the doubles count rate is often compared to doubles count rates of similar known standards. Through interpolation the mass of the measured sample can be estimated. The main disadvantage of this technique is that it requires known standards of similar composition and geometry, which may not always be available. Methods have been developed, the coupling method, to move from a known standards approach to one which incorporates first principle equations, however, these methods still rely on known standards [9][10]. The reason why the point-model equations do not work for active measurements is that with current measurement systems the point-model assumptions are violated. In particular, the assumption that the source of neutrons from the sample can be modelled as a point in space is not rigorously correct. Although neutrons with high energies can penetrate thick actinide materials, thus creating homogeneous fission rates within them, neutrons which have been slowed down to epithermal energies are easily absorbed in actinide materials due to capture and fission resonances. These fission resonances cause the majority of fissions to occur on the surface of the measurement sample, [11], thus creating a heterogeneous fission rate and violating the point-model assumptions.

One solution to the problems caused by epithermal neutrons is to prevent them from entering the measurement sample. This can be achieved by surrounding the measurement sample in high content $^{10}B$ material, such as boron carbide ($B_4C$). After ensuring that the fission rate in the sample is homogeneous the following first principle equations can be used:

$$(D_{Li} - D_{Passive}) = \frac{F_o {}^{239}Pu_{eff\_Li}\varepsilon^2 f_d M_L^2 \nu_{Li2}}{2}\left[1 + \frac{(M_L-1)\nu_{Li1}\nu_{Fis2}}{(\nu_{Fis1}-1)\nu_{Li2}}\right] \quad (1)$$

where $D_{Li}$ is the doubles count rate from the AmLi measurement, $D_{Passive}$ is the doubles count rate from the passive measurement, $^{239}Pu_{eff\_Li}$ is the effective mass of $^{239}$Pu in the sample that would produce the same AmLi doubles count rate as the complete sample, $\varepsilon$ is the detector efficiency, $f_d$ is the doubles gate fraction, $M_L$ is the leakage self-multiplication of the item, $\nu_{Li1}$ is the first moment of induced fission for neutrons with an AmLi energy spectrum, $\nu_{Li2}$ is the second moment of induced fission for neutrons with an AmLi energy spectrum, $\nu_{Fis1}$ is the first moment of induced fission for neutrons with a fission energy spectrum, and $\nu_{Fis2}$ is the second moment of induced fission for neutrons with a fission energy spectrum.

Because $F_o$, the specific fission rate, is based on the induced fission rate and not the spontaneous fission rate, it cannot be treated as a constant. $F_o$ can instead be solved for using:

$$\mathrm{F_o} = \frac{\varphi_{\mathrm{Li}}\sigma_{\mathrm{f\_Li}}\mathrm{N_A}}{\mathrm{M_{molar}}} \quad (2)$$

where $\phi_{Li}$ is the neutron flux within the sample during an AmLi measurement, $\sigma_{f\_Li}$ is the average $^{239}$Pu fission cross section for neutrons with an AmLi energy spectrum, $N_A$ is Avogadro's Number, and $M_{molar}$ is the molar mass of the effective mass isotope (approximately 239 amu in this example).

The neutron flux, $\phi_{Li}$, and the average fission cross section, $\sigma_{f\_Li}$, can be approximated as constants and determined through computer simulations. Because these values are fairly independent of the sample being measured, a generic geometry and composition sample may be used in the simulations.

Similar to the point-model equations, Eq. (1) is not directly useable. By combining Eq. (1) and (2) and rearranging terms we acquire:

$$^{239}\mathrm{Pu}_{\mathrm{eff\_Li}} = \frac{2(\mathrm{D_{Li}} - \mathrm{D_{Passive}})\mathrm{M_{molar}}}{\varphi_{\mathrm{Li}}\sigma_{\mathrm{f\_Li}}\mathrm{N_A}\varepsilon^2\mathrm{f_d}\mathrm{M_L^2}\nu_{\mathrm{Li2}}\left[1+\frac{(\mathrm{M}-1)\nu_{\mathrm{Li1}}\nu_{\mathrm{Fis2}}}{(\nu_{\mathrm{Fis1}}-1)\nu_{\mathrm{Li2}}}\right]} \quad (3)$$

Analogous to passive neutron coincidence counting, an equation relating the $^{239}Pu_{eff}$ mass to elemental masses can be created. This equation includes not just those isotopes of plutonium and uranium, but of neptunium and americium as well:

$$^{239}Pu_{eff_{Li}} = C_{U235_{Li}}m_{U235} + C_{U238_{Li}}m_{U238} + C_{Np237_{Li}}m_{Np237} + C_{Pu238_{Li}}m_{Pu238} + C_{Pu239_{Li}}m_{Pu239} + C_{Pu240_{Li}}m_{Pu240} + C_{Pu241_{Li}}m_{Pu241} + C_{Pu242_{Li}}m_{Pu242} + C_{Am241_{Li}}m_{Am241} + C_{Am243_{Li}}m_{Am243} \quad (4)$$

where $C_{k\_Li}$ is an equivalent worth constant for a AmLi measurement for isotope *k*, $m_{U235}$ is the mass of $^{235}$U in the sample, $m_{U238}$ is the mass of $^{238}$U in the sample, $m_{Np237}$ is the mass of $^{237}$Np in the sample, $m_{Pu238}$ is the mass of $^{238}$Pu in the sample, $m_{Pu239}$ is the mass of $^{239}$Pu in the sample, $m_{Pu240}$ is the mass of $^{240}$Pu in the sample, $m_{Pu241}$ is the mass of $^{241}$Pu in the sample, $m_{Pu242}$ is the mass of $^{242}$Pu in the sample, $m_{Am241}$ is the mass of $^{241}$Am in the sample, $m_{Am243}$ is the mass of $^{243}$Am in the sample.

The constants, $C_{k\_Li}$, can be determined by using a ratio of nuclear properties of the effective isotope [12]:

$$\mathrm{C_{k\_Li}} = \frac{(\sigma_{\mathrm{f\_Li}}\nu_{\mathrm{Li2}})_{\mathrm{k}}}{(\sigma_{\mathrm{f\_Li}}\nu_{\mathrm{Li2}})_{239}}\left(\frac{\mathrm{M_{molar,239}}}{\mathrm{M_{molar,k}}}\right) \quad (5)$$

where *k* represents the isotope of interest.

## 3. Measurements and simulations

In order to test the first principle measurement method a series of measurement campaigns were conducted at the Joint Research Centre (JRC) in Ispra, Italy and Los Alamos National Laboratory (LANL) in New Mexico, USA. These campaigns included measurements with the Active Well Coincidence Counter (AWCC) [13] and Epithermal Neutron Multiplicity Counter (ENMC) [14][15] in both passive and active modes. Americium-beryllium (AmBe), plutonium-boron (PuB), and AmLi were all considered as (α,n) interrogation sources, with AmBe being rejected due to its large internal (n,2n) reaction rate. A wide variety of nuclear materials were measured including: metals, powders, and ceramics; masses in the range of 5g – 170g; and isotopic compositions ranging from depleted uranium to High Enriched Uranium (HEU) and reactor grade to weapons grade plutonium [16-21]. Each measurement consisted of 60 – 90 cycles at 60 seconds per cycle. The detector response for each measurement was analyzed using the International Atomic Energy Agency (IAEA) International Neutron Coincidence Counting (INCC) software program in rates-only mode to determine the singles, doubles, and triples count rates with statistical uncertainties [22][23]. Each measurement performed was simulated in Monte Carlo N Particle Extended (MCNPX) to validate the MCNPX models. The AmBe neutron spectrum used was that of Cody Peeples [24], the PuB spectrum was determined from Sources4C [25], and the AmLi neutron spectrum was that of Geiger and Van der Zwan [26]. The number of histories run in each MCNPX simulation varied from 20 MegaHistories (20 MH or 20 million active histories) to 300 MH, depending on the measurement type being modelled. Both F4 and F8 tallies were used to determine the neutron flux in the measurement sample and the correlated count rates in the $^3$He tubes.

In order to prevent epithermal neutrons from entering the measurement sample, a $B_4C$ cylinder was manufacture with two cylindrical plates for the top and bottom. The $B_4C$ had dimension of an inner diameter of 13 cm, inner height of 17 cm, and a thickness of 2.5 cm on all sides. These dimensions were chosen based on (1) the size of the ENMC measurement cavity, (2) size of the nuclear standards to be measured, and (3) the epithermal absorption capabilities of the $B_4C$. The boron isotopic composition was that of natural boron.

Measurements of various plutonium standards, shown in Table 1, were performed using the ENMC in a passive, active AmLi, and active PuB mode. The $^{239}Pu_{eff}$ values calculated from Eq. 3 and 4 were determined for the measured, simulated, and declared data.

| Item | Pu [g] | $^{238}$Pu [wt.%] | $^{239}$Pu [wt.%] | $^{240}$Pu [wt.%] | $^{241}$Pu [wt.%] | $^{242}$Pu [wt.%] | $^{241}$Am [g] |
|---|---|---|---|---|---|---|---|
| STDIS03 | 10.96 | 0.0053 | 96.3504 | 3.5566 | 0.0328 | 0.0182 | 0.010 |
| STDIS09 | 11.85 | 0.0175 | 92.7957 | 6.8863 | 0.1216 | 0.0734 | 0.036 |
| STDIS12 | 20.09 | 0.0481 | 87.4464 | 11.8503 | 0.2787 | 0.2228 | 0.158 |
| CBNM-Pu61 | 5.45 | 1.0319 | 65.7930 | 26.6912 | 2.0697 | 4.4143 | 0.344 |
| STD11 | 59.74 | 0.0228 | 92.7048 | 6.6156 | 0.0741 | 0.5827 | 0.202 |
| LAO250C10 | 59.36 | 0.0470 | 82.9792 | 16.3044 | 0.3318 | 0.3376 | 0.621 |

**Table 1:** Mass and isotopic composition of the plutonium standards measured at LANL.

The AmLi and PuB results had similar trends, with excellent agreement between the simulated and declared, shown in Figures 1 and 2. This implies that the assay methodology could be both accurate and precise. The results from the measured data are harder to interpret due to the large measurement uncertainties. The large uncertainties come from Eq. 3 in which the passive and active doubles count rates are subtracted from each other. Because these two count rates are of similar magnitude their resultant subtraction is a small value with a large uncertainty. Performing longer measurements is not an ideal practical solution since this data was acquired with 1 hour measurements of the samples and overnight measurements of the background. A potential solution would be to increase the difference between these two count rates by increasing the AmLi source strength. Active measurements which have a large background source, such as the spontaneous fission neutrons from plutonium, can benefit from a stronger interrogation source [13]. Simulations were performed to quantify this potential solution. The AmLi source strength was varied over the ranged of $5x10^3$ to $1x10^7$ n/s per AmLi source. The AmLi source strength used in the measurements was $5x10^4$ n/s for each AmLi source. From the results shown in Table 2, it can be seen that for the larger mass plutonium standards, STD11 and LAO250C10, there is a significant reduction in the expected $^{239}Pu_{eff\_Li}$ mass uncertainty [27] for active measurements using stronger AmLi interrogation sources. Simulations using a PuB source had similar results.

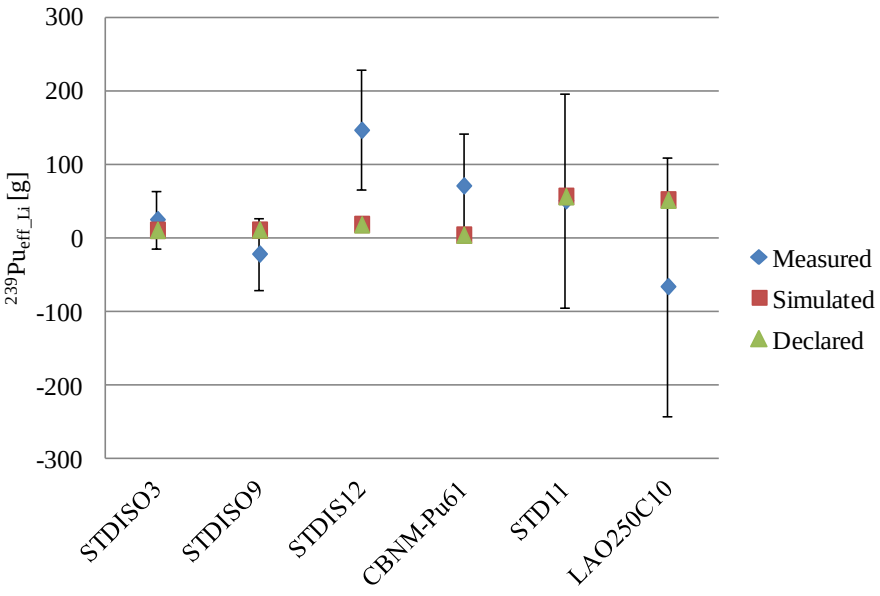


**Figure 1**. Measured, simulated, and declared $^{239}Pu_{eff\_Li}$ values.

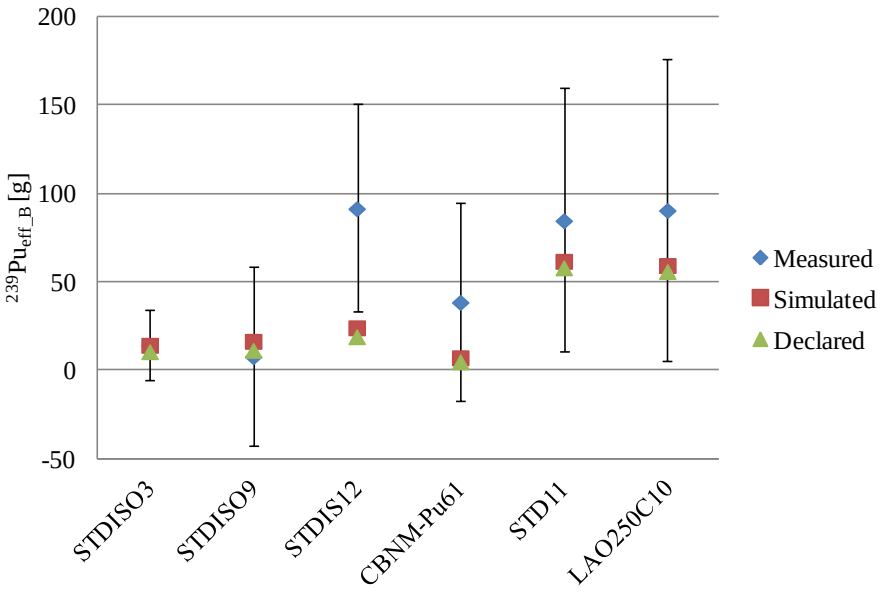


**Figure 2**. Measured, simulated, and declared $^{239}Pu_{eff\_B}$ values.

| Plutonium standards | AmLi strength [n/s each] | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | $5x10^3$ | $1x10^4$ | $5x10^4$ | $1x10^5$ | $5x10^5$ | $1x10^6$ | $5x10^6$ | $1x10^7$ |
| STDIS03 | 205.2 | 112.9 | 50.1 | 44.3 | 40.3 | 39.9 | 39.6 | 39.5 |
| STDIS09 | 336.7 | 176.0 | 61.0 | 49.3 | 41.2 | 40.3 | 39.6 | 39.5 |
| STDIS12 | 715.2 | 364.6 | 97.4 | 67.1 | 44.6 | 42.0 | 39.9 | 39.7 |
| CBNM-Pu61 | 636.8 | 326.6 | 90.0 | 63.4 | 43.6 | 41.3 | 39.5 | 39.3 |
| STD11 | 670.8 | 341.8 | 84.2 | 53.7 | 30.4 | 27.6 | 25.4 | 25.1 |
| LAO250C10 | 1233.2 | 624.9 | 141.8 | 82.8 | 36.4 | 30.7 | 26.0 | 25.4 |

**Table 2:** Source strength optimization for measured plutonium standards using the active configurations of the ENMC. Values shown are expected $^{239}Pu_{eff\_Li}$ [g] uncertainties.

## 4. Results of simulated samples

In order to test the full measurement methodology, simulated samples containing various actinide mixtures were created using Oak Ridge Isotope Generation and Depletion Automatic Rapid Processing (ORIGEN-ARP) Code and their neutron signatures determined using the ENMC MCNPX model. These samples can be organized into three categories: uranium extraction (UREX) type samples which contain U, Np, Pu, and Am at used fuel isotopic compositions [1]; MOX type samples which contain depleted uranium and weapons grade plutonium [28]; and uranium samples of varying enrichments. Each sample, seen in Table 3, consisted of 212.5 g of heavy metal mass at a density of 2.5 g/cc in a dioxide matrix. The sample geometry is a cylindrical shape (D=4.5 cm, H=6.288 cm) located inside a thin aluminium can of thickness 0.6 cm. The nomenclature on the sample names for the UREX and MOX samples corresponds to their isotopic composition (UREX or MOX) and their non-uranium heavy metal fraction (Np, Pu, and Am). The uranium samples are designated by the letter "U" followed by the $^{235}U$ isotopic fraction. All the MCNPX simulations were run for 250 MH, with the exception of spontaneous fission simulations which were run for 790 MH.

| Sample name | Uranium mass [g] | Neptunium mass [g] | Plutonium mass [g] | Americium mass [g] | $^{235}U^{(a)}$ fraction [%] |
|---|---|---|---|---|---|
| UREX1.27 | 210 | 0.14 | 2.46 | 0.10 | 0.90 |
| UREX50 | 106 | 5.61 | 96.60 | 4.04 | 0.90 |
| UREX10 | 191 | 1.12 | 19.32 | 0.81 | 0.90 |
| UREX10_NoNp | 191 | 0.00 | 20.40 | 0.85 | 0.90 |
| MOX50 | 106 | 0.00 | 106.25 | 0.00 | 0.25 |
| MOX5 | 204 | 0.00 | 10.22 | 0.00 | 0.25 |
| U93 | 213 | 0.00 | 0.00 | 0.00 | 93.00 |
| U50 | 213 | 0.00 | 0.00 | 0.00 | 50.00 |
| U20 | 213 | 0.00 | 0.00 | 0.00 | 20.00 |
| U5 | 213 | 0.00 | 0.00 | 0.00 | 5.00 |
| U0.72 | 213 | 0.00 | 0.00 | 0.00 | 0.72 |
| U0.25 | 213 | 0.00 | 0.00 | 0.00 | 0.25 |

**Table 3:** Simulated samples along with their name, actinide masses, and $^{235}U$ isotopic fraction. The $^{235}U$ fraction is relative to total uranium mass.

## 4.1. UREX and MOX results

Simulated assayed mass estimates for the plutonium and americium were within ±7% of the starting (exact) values. The $^{235}U$ and neptunium results, shown in Figures 3 and 4, were less accurate. Although the difference between the assayed and exact values is often good, it is difficult to draw conclusions from the data due to the large uncertainties. In particular, UREX50 has big uncertainties due to the relatively large amount of reactor grade plutonium in it, which creates a high doubles count rate. It is suspected by the authors that the uncertainty estimates are larger than they statistically should be, although the exact cause of the inflated uncertainties is not yet known.

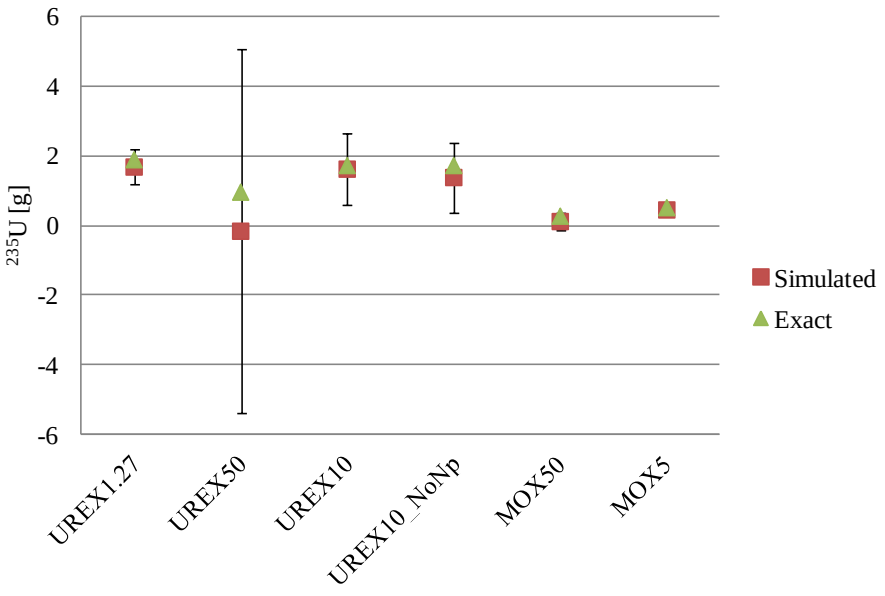


**Figure 3**. Simulated and exact $^{235}U$ masses of simulated UREX and MOX samples. The error bars shown are due to statistical uncertainties in the MCNPX results.

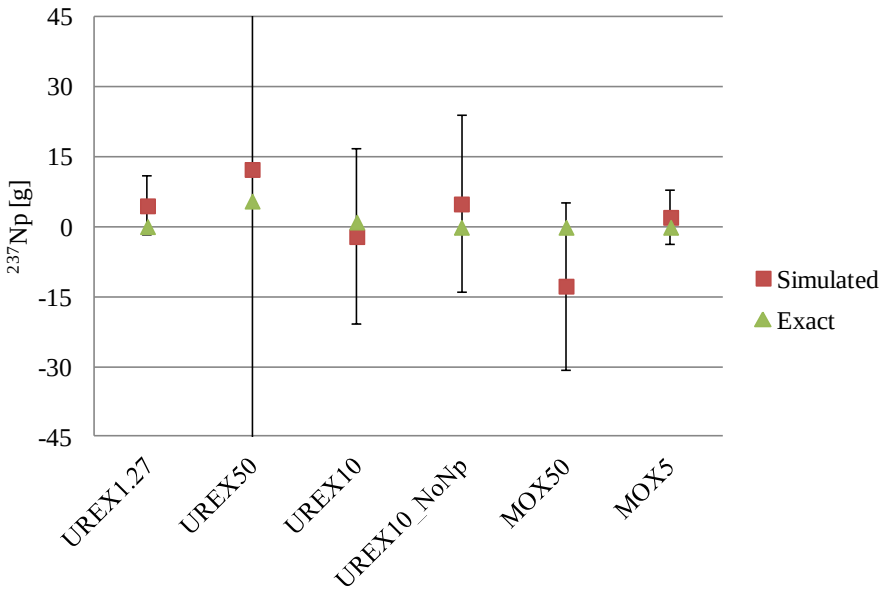


**Figure 4**. Simulated and exact $^{237}$Np masses of simulated UREX and MOX samples. The error bars shown are due to statistical uncertainties in the MCNPX results.

### 4.2. Uranium results

The $^{235}$U assay results, shown in Figure 5, matched very well with the exact values for the uranium samples. The percent differences are approximately -3% for the HEU samples and more negative for the low enriched uranium (LEU) samples.

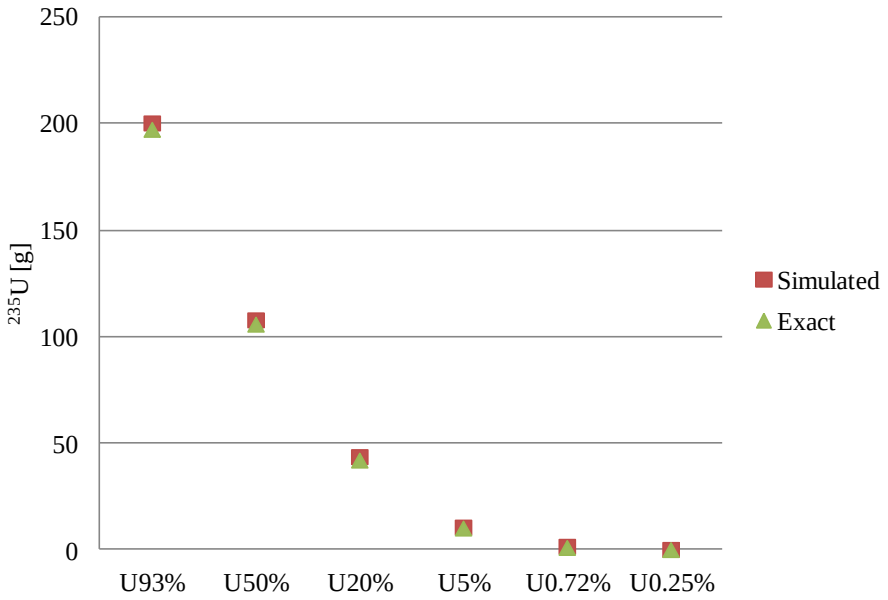


**Figure 5**. Simulated and exact $^{235}$U masses of simulated uranium samples. The uncertainties are too small to be plotted.

## 5. Discussion and conclusions

Current non-destructive assay (NDA) techniques have been developed to quantify uranium, plutonium, or MOX, but not samples which contain more than these two actinides. There are applications of more complex fuel materials, including neptunium and americium, which are currently being

developed. It is important to develop a method of quantitatively measuring these more complex materials before they become commercially common.

The approach taken to quantify a mixture of several actinides was to use a first principle neutron multiplicity counting approach. This technique has the added bonus of being applicable to measurements of shielded or heterogeneous MOX and uranium samples, as well as potentially weapons verification. It is postulated that weapons verification would be possible by analyzing the $^{240}Pu_{eff}$ and $^{239}Pu_{eff}$ values. These values can distinguish weapons useable materials from fission sources without revealing detailed information about the weapon. The required modifications to existing neutron multiplicity counters to allow for first principle methods to be applied is relatively minor and inexpensive, consisting of a cylindrical can of natural $B_4C$ and a high-energy (α,n) neutron source.

The main challenge of this technique is the ability to make precise measurements. As seen in the measured and simulated data, the slight differences between the passive and active doubles count rates can cause the results using current detector technology to be statistically irrelevant. Potential solutions to this challenge exist, such as more intense interrogation sources, but they have not been thoroughly tested.

The ability to quantify mixed actinide materials would currently be useful, and in the future may be a necessity. Further research is needed in developing this technique, in particular the ability to reduce uncertainties. Other aspects of this research should be investigated as well, such as results of kilogram sized, non-powder, non-oxide, and varying isotopic composition samples. It is suspected by the authors that this technique's full capabilities can only be utilized with advanced neutron multiplicity detectors, such as future fast neutron detectors.

## 6. Acknowledgements

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# JRC CBRN activities: Contribution to the Implementation of the EU CBRN Action and to the CBRN Centres of Excellence Initiative

S. Abousahl* and Z. Palajova
European Commission
Joint Research Centre
Brussels

said.abousahl@ec.europa.eu

## Abstract

Within the international security context, the Prevention and mitigation of Chemical, Biological, Radiological and Nuclear (CBRN) hazards has become an important area of activities. EU is contributing to the international community effort to mitigate the threat and risks associated with these materials.

The EU CBRN Action Plan (AP) was adopted in December 2009. The AP aims to strengthen CBRN security throughout the EU. Based on an all-hazard approach, the AP's overall goal is to reduce the threat of, and damage from CBRN incidents of accidental, natural and intentional origin, including terrorist acts. The AP contributes to the implementation of the EU Counter Terrorism Strategy and provides for three main areas of CBRN security work: Prevention, Detection, Preparedness and response. A total of 124 actions are to be implemented by the EU Member States and the EU Institutions. In addition to 25 actions relating to radiological and nuclear security, there are 32 actions covering biological or chemical security. A further 67 actions are horizontal actions in the sense that they apply to more than one area. The implementation of the Action Plan is guided by consultation with national authorities and other relevant stakeholders. The IAEA, Interpol, and Europol are closely associated to the implementation of the Action Plan.
Outside EU border, the Instrument for Stability in its Indicative Programme 2009-2011 identifies “capacity building against CBRN threats” as a necessary condition for risk mitigation and preparedness. Effective response to CBRN risk depends on cooperation and co-ordination between all levels of government, response organizations and international partners. An effective strategy to deal with the CBRN threat requires a very high level of co-operation and co-ordination among many different authorities within and among countries. Therefore, the EU launched in 2010 the 'EU CBRN risk mitigation – Centres of Excellence initiative'. The overall scope of the initiative is hence to strengthen the long-term national and regional capabilities of responsible authorities and administrative infrastructures, and develop a durable cooperation legacy in the fight against the CBRN threat.
JRC is supporting the EC DG DEVCO in the technical implementation of the CBRN CoE initiative through activities such as the analysis of CBRN needs and project proposals of partner country/region, Preparation of terms of references for selected projects, follow-up and quality control of projects, management of the IT tools (website, databases); as well as communication and coordination between all stakeholders.

The JRC has long-standing experience and expertise in the field of nuclear security and safeguards in support to EURATOM, the IAEA and to implementation of EU projects under

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TACIS (Technical Assistance to Commonwealth of Independent States), Instrument for Nuclear Safety Cooperation and Instrument for Stability. It was logical that JRC have been attributed several RN actions of the EU CBRN AP and also the technical implementation of the CBRN CoE Initiative.

In this paper, we will highlight some of the activities of JRC in support to the implementation of the EU CBRN AP as well as the role and activities of JRC under the CBRN CoE initiative.

## 1- Introduction

Within the international security context, the Prevention and mitigation of Chemical, Biological, Radiological and Nuclear (CBRN) hazards has become an important area of activities. EU is contributing to the international community effort to mitigate the threat and risks associated with these materials.

Inside the EU, Member States and the commission worked on an EU CBRN Action Plan (AP) that has been adopted by the EU council in December 2009. The AP aims to strengthen CBRN security throughout the EU. Based on an all-hazard approach, the AP's overall goal is to reduce the threat of, and damage from CBRN incidents of accidental, natural and intentional origin, including terrorist acts. The AP contributes to the implementation of the EU Counter Terrorism Strategy and provides for three main areas of CBRN security work: Prevention, Detection, Preparedness and response. A total of 124 actions are to be implemented by the EU Member States and the EU Institutions. In addition to 25 actions relating to radiological and nuclear security, there are 32 actions covering biological or chemical security. A further 67 actions are horizontal actions in the sense that they apply to more than one area. The implementation of the AP is guided by consultation with national authorities and other relevant stakeholders. The IAEA, Interpol, and Europol are closely associated to the implementation of the Action AP.
Outside EU, the Instrument for Stability in its Indicative Programme 2009-2011 identifies “capacity building against CBRN threats” as a necessary condition for risk mitigation and preparedness. Effective response to CBRN risk depends on cooperation and co-ordination between all levels of government, response organizations and international partners. An effective strategy to deal with the CBRN threat requires a very high level of co-operation and co-ordination among many different authorities within and among countries. Therefore, the EU launched in 2010 the 'EU CBRN risk mitigation – Centres of Excellence initiative'. The overall scope of the initiative is hence to strengthen the long-term national and regional capabilities of responsible authorities and administrative infrastructures, and develop a durable cooperation legacy in the fight against the CBRN threat.

As the Commission's in-house science service and the only Commission service responsible for carrying out direct research, the mission of the JRC is to provide independent advice and evidence-based scientific and technical support throughout the whole policy cycle. The JRC security activities are tailored towards supporting the implementation of EU policies and strategies in the field such EU security strategies and related international treaties, energy, trade, customs, industry and home affairs.

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In the field of nuclear and radiological security, the JRC objectives are well defined and aim to ensure in one hand EU with an efficient and effective safeguarding and enhanced proliferation resistance of current, innovative and future nuclear fuel systems and in another hand to support the EU and its Member States in enhancing its internal and external security by combating illicit trafficking of nuclear and radioactive materials in the areas of prevention, detection and response including nuclear forensics.

The JRC has long-standing experience and expertise in the field of nuclear security and safeguards in support to EURATOM, the IAEA and to implementation of EU projects under TACIS (Technical Assistance to Commonwealth of Independent States), Instrument for Nuclear Safety Cooperation and Instrument for Stability. It was logical that JRC have been attributed several RN actions of the EU CBRN AP and also the technical implementation of the CBRN CoE Initiative.

In the following, we will highlight some of the activities of JRC in support to the implementation of the EU CBRN AP as well as the role and activities of JRC under the CBRN CoE initiative.

# 2 JRC Contributions to the Implementation of EU CBRN AP

## 2.1 Nuclear Security Training

### 2.1.1 Establishing EU Nuclear Security Training Centre (EUSECTRA)

With the objective to establish a security training programme applicable to the law enforcement community, the European Nuclear Security Training Centre (EUSECRA) project was launched in 2011 based on the findings of the feasibility study achieved in 2010.
Training areas for EUSECTRA include for the most part border detection, train-the-trainers, mobile emergency response, reach-back, creation of national response plans, nuclear forensics, radiological crime scene management, nuclear security awareness and sustainability of a national nuclear security posture. EUSECTRA facilities are established at the JRC sites in Karlsruhe and Ispra (Italy).

The project was designed to raise the security awareness and culture of law enforcement staff and in particular front line officers in order to successfully deal with threats involving nuclear and other radioactive materials.
Training areas for EUSECTRA include mainly border detection, train-the-trainers, mobile emergency response, reach-back, creation of national response plans, nuclear forensics, radiological crime scene management, nuclear security awareness and sustainability of a national nuclear security posture.
Number of activities was carried out to strengthening the EUSECTRA technical infrastructure. These covered (see fig 1):

- Establishment of outside and an inside (Caisson) training area at the JRC Karlsruhe
- Upgrading of measurement and detection equipment on both sites of JRC Karlsruhe and JRC Ispra.

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Significant efforts have been granted to deliver the EUSECTRA an extensive range of equipment and apparatus to cover EU Member States and anticipated international customers commonly deployed and operated gears: personal radiation detectors, hand held survey meters, radioisotope identifier devices, radiation portal monitors etc.

**Fig. 1:** Drawing of the EUSECTRA outside training area

Moreover, all EU Member States were requested to identify experts who could contribute to the training courses. Training material on detection (front-line officers and train-the-trainers), management of radiological crime scenes, development of a National Response Plan, reach back and measurement expert support team (MEST), nuclear forensics etc. is continually updated in the light of the experience gathered during training sessions.
First training sessions served for further optimization of the syllabi, for early identification of areas of improvement and for establishing strong relations with EU Member States and international expert partners.
On 18 April 2013, official inauguration of EUSECTRA took place. EUSECTRA will continue to focus its efforts to ensure sustainability of the nuclear security training in partnership with EU Member States and international customers.

### 2.1.2 Setting up a European virtual reality based training capacity (EVIRETRAC) in the area of detection of nuclear and radioactive materials

This project has started this year and aims to develop Virtual Reality based tools for training first responders in nuclear security scenarios. It is intended as a complement to, not replacement of, hands-on training as developed within the EUSECTRA. The project aims at the development and adaptation of the current syllabus, as used in EUSECTRA, to the use in EVIRETRAC. It includes a design of a concept for an “ideal” VR based nuclear security training room and Build a prototype of the above referred “pilot concept” for a VR based nuclear security training room.
JRC will team up with organization(s) with recognized experience in the field. In order to get significant results in the near term, more than developing something completely new from scratch, JRC will look as much as possible to existing tools incorporating already some of the required capabilities (f.i. tools for radiation protection applications) and contract the developer(s) of such tools in order to modify them according to the needs for nuclear security training.
In addition to the simple non-immersive tool (similar to the classical videogames), JRC will also start to implement a first approach towards totally immersive training by developing a

prototype of CAVE approach (Cave Automatic Virtual Environment) where the trainee can experience a full immersion in a virtual world.

## 2.2 Testing detection technologies

### 2.2.1 Illicit Trafficking Radiation Assessment Program: ITRAP +10

Ten years after the first ITRAP project (1998-2000) carried out by Austria and IAEA, the Joint Research Centre and the US Department of Homeland Security (DHS-DNDO), in cooperation with the International Atomic Energy Agency (IAEA) and the US Department of Energy (DoE), are jointly implementing the Illicit Trafficking Radiation Assessment Program+10 (ITRAP+10).
The ITRAP+10 project carries out an evaluation and comparison of the performance of available radiation detection equipment relevant to nuclear security. The results provide an independent assessment of the available radiation detection equipment on the market which will serve as a reference for regulatory and other Member State authorities to identify equipment and or families of equipment to address their particular needs, and help to ensure common standards at a European level.
The certification of border monitoring instruments itself is not part of the project goals, however the project's results may help the European Commission to start establishing a network of European laboratories able to perform a variety of tests included in the referenced standards.
Tests carried out focused on the nuclear/radiological performances of the border monitoring instruments and are mainly based on the technical specifications described in the corresponding European and/or International standards.
Eight different families of equipment were tested:

- Radiation Portal Monitors (RPM) for Road Vehicles
- Spectrometric Radiation Portal Monitors (SRPM)
- Personal Radiation Detectors (PRD)
- Spectrometric Personal Radiation Detectors (SPRD) (see fig 2)
- Radioisotope Identifier (RID)
- Highly sensitive Gamma Search Detectors (GSD)
- Highly sensitive Neutron Search Detectors (NSD)
- Portable Radiation Scanners – Backpack type (PRS)

with approximately 100 instruments in total for both EU and US laboratories.
The outcome of this project will be a final report on the current technical level of commercially available equipment used in nuclear security and it will define the general performance capabilities of each equipment class. The project has entered its final phase and manufacturers are receiving a personalized test report containing a ranking of the instruments tested done parameter by parameter (or test by test).

Polimaster Saphymo Aspect Symetrica

**Figure 2:** Radiation Portal Monitors (RPMs) tested within ITRAP+10 in EU

### 2.2.2 Assessment of the influence of weather conditions on the detection capabilities of a standard Radiation Portal Monitor for radioactive/nuclear material

However, the testing and evaluation are done at laboratory scale and the influence of the environment conditions on the performance of the equipment is missing. This missing piece of information will be covered under a new project with the objective of this study is to determine what could be the effects of a (heavy) rain on the capability of a vehicle radiation portal monitor (RPM) to detect the presence of nuclear/radioactive material or to cause false alarms. Since this type of equipment is typically installed outdoor at border crossing points, the environmental conditions might influence their performances and affect the detection efficiency. Once the influencing parameters and their effect will be identified, corrective actions could be investigated and implemented.

## 2.3 Support to IAEA Illicit Trafficking Data Base

JRC carries out a detailed study on the potential areas of improvement of the IAEA's Illicit Trafficking Database (ITDB) according to the recommendations in the EU CBRN Action Plan and proposes solutions and suggestions for improvements, and implements them when possible in the frame of the resources.
The ITDB, established in 1995, collects information on incidents, reported by national Points of Contact (PoC) involving unauthorized acquisition, provision, possession, use, transfer or disposal of nuclear and other radioactive materials, whether intentionally or unintentionally.
Through this project, JRC addresses two main concerns related to the ITDB, namely:

- Is the information provided sufficient for the users of the system?
- How to ensure real time accessibility to the information for law enforcement authorities?

Input from the EU-27 Member States was sought through organizing a workshop with all stakeholders concerned and through a questionnaire concerning suggestions of the EU-27 for improvements of the ITDB.
Based on the findings obtained from the questionnaire, workshop and from the national PoC meeting in Vienna, the main areas for improvements were identified. In two of these areas,

JRC decided to take further steps: Web-ITDB, which would improve the data submission in many ways and Data security of the web-ITDB.
Currently, JRC continues to implement improvements/evaluate their practical feasibility.

### 2.3 Assessment and validation of modelling tools and decision support system

In addition, JRC carries out project focusing on assessment and validation of existing modelling tools that are or can be used in decision support systems following RN incidents.
It is foreseen that this objective will be achieved by preparing, sending out and evaluating an exhaustive questionnaire to EU-27 National Authorities competent for CBRN emergency response. Results of this study will be presented and discussed with all Member State Authorities involved during an international workshop, which will also serve as a platform to collect comments and ideas to draft recommendations for a strategy paper that could serve as a basis for a multi-annual work plan.

### 2.4 Nuclear Forensics

Finally, under a new project JRC will extend its support to Member States in the field of nuclear forensics. Legal arrangement will be established for operational nuclear forensics support to MS for the basic characterisation of intercepted nuclear material, which needs to be followed by an advanced nuclear forensic investigation at JRC Institute for Transuranium Elements (ITU).
Based on a questionnaire on the current nuclear forensics capabilities of the EU-27 Member States, syllabus for a training course on core capabilities in nuclear forensics will be prepared and appropriate Member States will be invited to a training course. This aims at establishing core capabilities in nuclear forensics in all EU Member States, recognizing that few Member States do have capabilities which go beyond that.

As investigations on illicit trafficking of nuclear materials may require analytical information that cannot be acquired by core capabilities, legal basis for operational nuclear forensic support (i.e. analysing the samples at ITU) needs to be in place. Therefore, it is intended to offer concluding a collaboration agreement (or equivalent) with interested Member States. Lastly, the whole procedure, including transport of "unknown" nuclear material and its analysis will be exercised in so-called joint analysis exercise.

## 3 JRC support to the implementation of the EU CBRN Centres of Excellence initiative

### 3.1 Background

The EU CBRN Risk Mitigation Centres of Excellence Initiative, launched in 2010 under the Instrument for Stability and with a budget of Euro 100 M€ for 2009-2013, aims at implementing a comprehensive strategy for reducing national and international vulnerability to CBRN risks. The origin of these risks can be criminal, accidental or natural. The initiative is designed by the European Commission and the Eurpean External Action Services EEAS. It is managed by EC DG DEVCO (European Commission Directorate General Development

and Cooperation) and implemented by the JRC and the United Nations Interregional Crime and Justice Research Institute (UNICRI) with a consortium leaded by ESSEC in charge of the development and implementation of governance principles. The initiative is developed with the technical support of relevant International and Regional Organisations, the EU Member States and other stakeholders. The initiative focuses on strengthening institutional capabilities of the partner countries to mitigate CBRN related risks and to support multilateral cooperation. This is achieved also through implementation of tailored projects developed by partner countries based on their need assessment.
Technical scope of projects varies from awareness raising, strengthening legislative framework and regulatory infrastructure, provision of training, and detection equipment, development of response plans, sharing best practice etc. as for example project no. 28 focuses on supporting development of an integrated national security system for nuclear and radioactive materials in South East Asia.
For the time being, the initiative is implemented in 37 official CoE partner countries (on-going negotiations with many potential partner countries) grouped around 8 CoE Regional Secretariats:

- African Atlantic Façade
- Central Asia
- Eastern and Central Africa
- Gulf Cooperation Council Countries
- Middle East
- North Africa
- South East Asia
- South East Europe, Southern Caucasus, Moldova and Ukraine

The partner countries are represented through National Teams of CBRN experts, composed of the National Focal Point and representatives of relevant national agencies/institutions, and Regional Secretariats (RS), which are the visible focal points of the CoE in the regions.

### 3.2 JRC activities in support to CBRN CoE

The activities of JRC in support to the implementation of the CoE initiative concern all the CoE cycle of activities including coordination, communication, monitoring and sustainability aspects

With regard to the number of stakeholders involved directly or indirectly in the CBRN CoE initiative the need for coordination of activities is eminent. As JRC is (except for DEVCO) the only European Commission DG involved in the implementation of the initiative, the coordination between DEVCO and JRC represents the first level of coordination. Inevitably, JRC needs to coordinate its activities with all the rest of the internal CBRN CoE stakeholders. Taking into account the importance of the Regional Secretariats in facilitating communication with partner countries, regular communication with regional coordinators is crucial for 'project cycle activities' and missions planning.

In addition, JRC coordinates with other services of the European Commission (mainly DG MOME Affairs) to achieve optimal results and ensure that the recommendation of the

European Parliament on the close link and mirror effect between CBRN security inside and outside EU is applied.

Furthermore, JRC ensures the coordination with international organisations as for example IAEA and its network of (national) 'Nuclear Security Support Centres (NSSC)' and other international partners, based on the experience of and through the Border Monitoring Working Group established in 2005 between EU, US and IAEA, International Export Control Capacity Building Working Group (XCCB-WG) launched recentely between EU and US, BioChem Working Group and Engagement of Scientists Working Group that are under establishment.

Information sharing and close cooperation with partner countries is essential for successful implementation of the initiative and a strong commitment from partner countries to actively participate in the initiative is of key importance. In this respect, JRC makes efforts to enhance partner countries engagement in the initiative among other things through supporting them in performing their tasks, active participation to Round Table and bilateral meetings, sharing information and collection of their feedback.

With the aim to promote the initiative and to facilitate the communication and coordination among stakeholders, a private portal was created. The CoE private portal developed by JRC contains databases on CBRN capabilities, information on projects, related documents, online agenda, etc. Moreover, it is and will play important role for knowledge management, as it is/will be used to archive documents and projects implemented within the CoE initiative and also by other donors within the CoE technical and geographical scope. In addition, quarterly newsletter is being published in English and Russian and gradually also in French and Arabic, public website has been created and promotional material is designed and disseminated by the JRC. The activities further focus on: ensuring communication between all stakeholders, expanding the CoE network of experts, collecting information about National Focal Points, National Teams and Regional Secretariat, promoting and presenting the CoE initiative, and ensuring the EU visibility.

The 'CoE cycle of activities' includes: CBRN guidelines development, need assessment and National Action Plan development, drafting, reviewing and ranking project proposals, drafting Terms of Reference, monitoring and quality control of projects.
In this respect, JRC works on finalizing the Need Assessment Tool (first version prepared by UNICRI) and its electronic version. JRC will also provide support to partner countries in drafting their National Action Plans as well as in drafting project proposals' when needed.

Moreover, JRC provides support to DEVCO in preliminary analysis of submitted project proposals, assistance in preparation of Terms of Reference for selected projects and provides other technical inputs to DEVCO upon request.

Apart from coordination of projects' implementation, JRC is responsible for monitoring and evaluation of some of the projects and thus for providing systematic, consistent and reliable information on projects' progress against objectives and performance targets laid down in their Terms of Reference. Through this activity a basis for decisions on necessary modifications for future projects will be provided as for example: resource utilization may be

adjusted, priorities shifted and new activities introduced. The ex-post evaluation will assess final results, impact and sustainability of implemented project.

On the top of that, JRC will continue supporting development of countries ownership through helping them in carrying out their own duties/responsibilities, creating relationships that foster cooperation, involving them in assessment and evaluation of projects and promoting their successes.
Through number of meetings, seminars and workshops, JRC will ensure the link between implementers, national bodies, academia, NGOs, think-tanks and other stakeholders and disseminate good practices and lessons learned during projects implementation.

## 4 Conclusions

In fully accordance with his mission JRC have been attributed for implementation several RN actions of the EU CBRN action Plan in the areas of detection technologies, forensics, response and trainings. Some of the action have been successfully accomplished and some are under implementation. JRC is supporting the EC DG DEVCO in the technical implementation of the CBRN CoE initiative through activities such as the analysis of CBRN needs and project proposals of partner country/region, Preparation of terms of references for selected projects, follow-up and quality control of projects, management of the IT tools (website, databases); as well as communication and coordination between all stakeholders.

# First activities of the new ESARDA Export Control Working Group

**Filippo Sevini[1], Sandro Zero[2], Quentin Michel[3]**
**on behalf of all contributors**

Institute for Transuranium elements
Joint Research Centre, European Commission
Via Fermi 2749, Ispra 21027, Italy
E-mail: filippo.sevini@jrc.ec.europa.eu

[2] AREVA,
1 Place Jean Miller
92084 Paris La Défense, France

[3] Etudes européennes / European Studies
Political Science Department
Faculty of Law and Political Science
Liege University
Boulevard du rectorat 7B31, boite 30
4000 Liège

***Abstract:***

*Export control of dual-use goods is an important barrier against the proliferation of weapons of mass destruction.*
*The new ESARDA working group on export controls was launched on October 15-16, 2012, following the recommendations of the Reflection Group 2010 and the interest arisen within the VTM WG, where a sub-Group on export control had been established in 2011, with a positive feed-back and paving the way to the new WG.*

*ESARDA EXP-WG has no political mandate, but compared to other groups dealing with the subject, presents the added value of including various types of contributors and stake-holders with different views and needs related to export control. The first meetings were attended by nuclear regulators; suppliers and operators; R&D organisations; universities; NGOS; IAEA and European Commission (JRC and ENER).*

*The planned research subjects encompass exchanges of experience on implementation issues, such as Intra-EU transfers or intangible technology transfers, as well as dual-use technology and items.*

*The paper summarises the background of the new WG, develops on the topics discussed in the first meetings and presents a view of the next steps.*

**Keywords:** export control, dual-use goods, sensitive trade, intangible technology transfers, non-proliferation, WMD, technology, compliance, Nuclear Suppliers Group

## 1. Introduction

The new Export Control Working Group (EXP-WG) was officially kicked-off on October 15-16, 2012 as result of a process started in 2006.
The first export control workshop within ESARDA had been organised by the Verification Technologies and Methodologies, VTM Working Group at JRC Ispra in October 2006 with large participation

including IAEA and US DOE. Following its positive outcome and an increasing attention over the years, VTM WG decided in May 2011 to start a dedicated sub-WG, which witnessed great participation and interest for a broad scope of activities.

On the occasion of its second meeting in January 2012, the sub-WG participants expressed the wish to propose the creation of a dedicated Working Group, dealing with the complex multi-disciplinary character of the subject.

## 2. EXP-WG's scope and working method

Care was taken to identify the real need for a new WG, knowing that various other international committees and groups are focused on non-proliferation and indirectly on export control.

Various formal, legally binding as well as informal working groups or committees focus on export control in the EU and elsewhere.

In the EU, the European Commission DG TRADE chairs the "Dual Use Coordination Group Art.23" sharing experiences on the implementation of the legal framework, while the Council's rotating Presidency chairs the "Working Party on Dual-Use goods", as part of the legislative co-decision process. These two committees are legally mandated and formed by EU-27 representatives.

ENEF (European Nuclear Energy Forum) set up by the Council and supported by DG ENER, is mostly attended by technology holders and suppliers. The Sub-group on non-proliferation holds focussed events on export control, but does not plan having a constant agenda on export control.

In the academic world, the Chaudfontaine Group devotes annual meetings to dual-use EU trade control legal issues, attended by young researchers.

It was hence felt useful to have a non-binding forum to discuss export control research, technical aspects as well as implementation issues and legal framework, taking advantage of a mixed and not-exhaustively representative membership.

The principal added value of ESARDA is indeed to include various types of stake-holders, such as nuclear regulators, technology holders and exporters, universities and research centres, European Commission and IAEA.

This mix allows having discussions, exchanges and analyses from various points of view at the same time, providing reports on common issues that could be later sent for information to the authorities in charge of laws and regulations, for their possible peruse.

Various topics of interest have been analysed and included in the Terms of Reference, published on www.esarda.eu, including:

- EU and international framework updates
- IAEA Additional Protocol
- Intra-EU transfers
- Internal Compliance Programmes
- Intangible Technology Transfers
- Dual-use technology

Besides meeting reports, the WG plans to produce Technical sheets and reports e.g. on topics like ITT and compliance for researchers.
During the first meetings, discussions on some of the above topics have been started, as briefly summarised hereinafter.

## 3. Export control background

Nuclear export control developed in parallel to nuclear safeguards, as reaction to the proliferation events.

The first steps started in the 50's with COCOM [1], during the Cold War, defining a first list of sensitive nuclear equipment. After the signature of the 1968 Non-Proliferation Treaty (NPT), the Zangger Committee was established, producing the first Trigger List, interpreting the NPT "*especially designed or prepared for*" concept.

The Indian test in 1974, revealed the necessity to convince suppliers countries not member of the NPT to adopt an export policy in line with the NPT principles. The Nuclear Suppliers Group (NSG) was established in this regard, adopting a list of sensitive items similar to the Zangger one.

The discovering in 1991 of the Iraqi nuclear weapons research program demonstrated that controlling only especially designed nuclear items was not sufficient. A list of dual-use items was therefore adopted in 1991.

Both NSG lists are published as IAEA INFCIRC 254, respectively Part 1 "Guidelines for Transfers of Nuclear-related Dual-use Equipment, Material, Software and Related Technology" and Part 2 [2,3]. Both lists are currently the object of a Fundamental Review process expected to conclude in 2013. Two first sets of amendments are included in "interim versions" of the lists available on NSG public web-site, maintained by the Joint Research Centre of the European Commission in collaboration with German BAFA. At the end of the review process all changes will be officially communicated to IAEA and issued as updated INFCIRC 254 Part 1 and Part 2.


**Fig. 1** - Nuclear Suppliers Group logo on www.nuclearsuppliersgroup.org

Nuclear export control regimes were later followed in the 80's by the Missile Technology Control Regime (MTCR), by the Australia Group (AG) for chemical and biological weapons and by the Wassenaar Arrangement in 1996 for dual-use items in general.

The EU decided in the 90's to integrate the control lists produced by the International Export Control Regimes and the Chemical Weapons Convention into a single one, the so-called "EU dual-use control list", included as Annex I to the dual-use regulation [4]. This initiative was very successful and provided model and inspiration to the the US EAR Commodity Control List, which has the same structure and coding, with about 25% more items. It is also gradually adopted by various non-EU countries.

## 4. EU and international framework

Called also for UNSCR 1540 [5], and rooted in EU's historical non-proliferation aims summarised by EU WMD strategy [6,7], the control of dual-use goods in the EU is ruled by EC Reg. 428/2009 and its amendments. The latest ones are EU Reg. 1232/2011 [8] and 388/2012 [9], which respectively contain new EU general export authorisations and an amended Annex I control list.

Regulations targeting specific countries may also include dual-use item, as it is the case of the recently amended Council Reg. 1263/2012, with target measures against Iran [10], DPRK [11] and the new Regulation 509/2012 with target measures against Syria [12].

Customs enforcement is ruled by the Community Custom Code Reg. 2913/92 [13]; its 2005 Security Amendments defined a common EU set of risk assessment criteria and communication platform [14].

The European Commission DG TRADE launched in 2011 a Green paper on the evolution of dual-use trade control [15]. Over 100 replies were summarised in a Commission Staff Working Paper [16] which will be presented to the European Council and Parliament.

This will pave the way to proposals for an evolving strategic trade control framework in the EU, which could include common risk criteria, structured catch-all controls and extended information exchanges.


**Fig. 2** – Pictorial view of the integrated EU Dual-use control list

## 5. IAEA Safeguards, Additional Protocol and export control

The discovery of Iraqi proliferation programme and the realisation of the limits to inspection activities triggered the definition of the 1997 Model Additional Protocol to the Safeguards Agreements with IAEA (INFCIRC 540) [17], which has enabled the Agency to gain access to a much wider range of information and locations.
The Additional Protocol includes also the requirement to provide export declarations for the nuclear items listed in Annex II (NSG Trigger List items) and also allows the Agency to request ad-hoc import data, to cross-check other declarations. 119 States had Additional Protocols in force on 9 April, 2013 [18].

EU member states usually rely on EURATOM to gather the declarations, and send them to IAEA in Vienna, depending on being or not "side letter" States. The data collection mechanism of exports and imports (if necessary) is quite important to the Agency in order to verify correctness and completeness of State declarations provided according to safeguards agreements. The collection of information allows the IAEA to perform a State level approach process which includes: information derived from obligatory State declarations and other reports; Agency's own verification activities (Short-Notice, Unannounced and Complementary Access inspections); satellite imagery, environmental sampling, forensics and open source information analysis.

Important indicators are trade patterns, procurement and R&D activities. These include also the analysis of industrial infrastructure to identify gaps, proliferation financing and indicators of incompleteness through trade data analysis [19, 20].

The State Level conclusions are summarised in the official IAEA Board of Governors reports flagging concerns and threats.

A paper on the experience of some ESARDA members with the Additional Protocol's export authorisations was presented at ESARDA 33$^{rd}$ symposium in Budapest [21]. The topic was later developed in a meeting of the new EXP-WG.

A problem stressed by ESARDA contributors is the difficulty to retrieve the data concerning the exports which actually took place. The export date does not coincide with the authorisation date and it is not possible to ascertain if an authorised export actually did take place. Customs data are therefore needed to verify the exports.

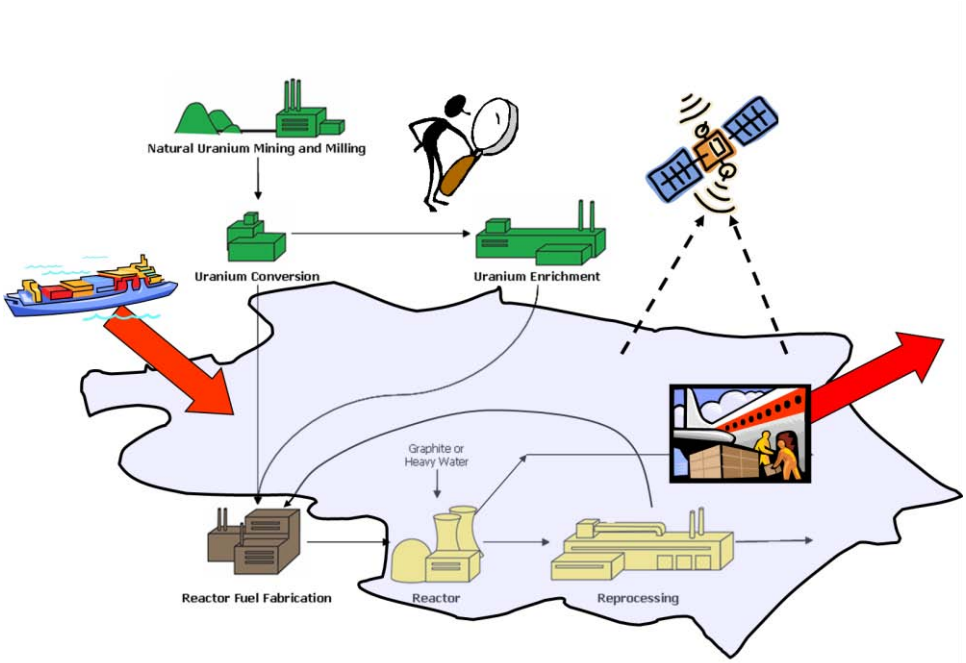


**Fig. 3** – Example picture summarising the various indicators taken into account by the State level approach

Another observation made was that AP Annex II contains 1995's rev. 2 version of NSG Trigger List (TL), which in the meantime has been updated several times. To facilitate coherence with export control laws, AP Annex II should be updated to the latest TL revision.

An extension of AP Annex II extension to include also NSG dual use items would offer IAEA more indicators, though still not exhaustive, and would also be coherent to the licensing requirements.

EXP- WG debated however that a bigger amount of information could also be redundant to the Agency and overload with data not always meaningful the tracing of undeclared activities. Indeed a report at the meeting showed that many declarations do not contain information useful to the analysts, in particular if related to intra-EU movements between large companies' branches. AP Annexes revisions are anyway not foreseen for the time being.

The discussion will be continued also in collaboration with the ESARDA Implementation of Safeguards WG; the results and collection of experiences will be summarised in a report.

## 6. Intra-EU transfers

Another topic addressed by the new WG is the transfer of controlled items within the EU.
Intra-EU exports are subject to EC Reg. 428/2009's Annex IV, which is divided in two parts:.

**Part I**
Includes items for which National General Authorisations may be defined:
- Items of stealth technology
- Items of the Community strategic control
- Items of the Category 5 Part 2 — Cryptography
- Items of the MTCR technology:

**Part II**
Includes items for which National General Authorisations are not allowed, i.e.:
- Items of Chemical Weapons Convention technology
- Items of NSG technology
  - All Annex I's Category 0 (=*NSG Trigger List*), subject to the following:
    - 0C001 (natural uranium, depleted uranium, thorium) is not included

- 0C002: only separated plutonium; uranium enriched in U235, U233 >20% are included

- Some dual Use items included: tritium production facilities, lithium production facilities, Electromagnetic Isotope Separation, neutron generators, cameras, pressure sensors

Even if related to dual-use missile technology and control equipment, *Annex IV does not control the items of the MTCR technology that are transferred on the basis of orders pursuant to a contractual relationship placed by the European Space Agency (ESA) or that are transferred by ESA to accomplish its official tasks;*

Large companies based on more EU countries have to comply with intra-EU transfers whose authorisation process is currently practically the same as a normal export from the EU.

This applies also to nuclear suppliers, despite the existence of the EURATOM Treaty. This fact has historical reasons: after the signature of the Non Proliferation Treaty and more specifically the establishment of the NSG, some EU member states felt that the nuclear common market was not applicable in view of their individual commitments. This understanding was not shared by other Member States, who were referring to the EU declaration relating to NSG Guidelines adhesion stating that NSG commitments shall be apply without prejudice of the EU treaties.

In 1984 a compromise was found with the Declaration of Common Policy adopted by the foreign ministers of the Community (Dublin Declaration, [22]), which introduced the transfer notification (not authorization) principle for exchanges between Member States, covering in particular arrangements for the transfer between Member States of separated plutonium and uranium enriched to more than 20% in the isotopes 235 or 233 and the transfer of installations, principal components of crucial importance, reprocessing or enrichment technology or the technology of heavy water.

The implementation of the Single Market (January 1993) opened *de facto* the free movement of dual-use items within the EC. It was hence necessary either to harmonise the national export control regimes or to adopt a single export control regime. No consensus was reached among MS.

A first regime constituted by two instruments – Regulation 3381/94 and a Council Joint Action 94/942/CFSP - was therefore adopted in 1994. This Regime established common principles and left to Member States Authorities the competence to decide and grant authorisation. This dual system was invalidated by the European Court of Justice and replaced by a single Council Regulation 1334/2000 (later recast as 428/2009) organised the export control of dual-use items. Nevertheless the division of competence between EU and its Member States remained unchanged.

The resulting Annex IV of Regulation 428/2009 ruling and controlling intra-EU transfers is a compilation of the previous Annexes IV, V and the Dublin Declaration. The NSG Trigger list (Cat. 0) is fully included, with some exceptions.

Intra-EU transfer controls impose a big burden on EU suppliers / exporters who have to request many licenses. As an example, up to several hundred authorisations seem to be needed for a nuclear reactor exported from France to Finland, whereas US competitors can do the same with one license only.

A first discussion held in the WG led to conclude that the current intra-EU transfer requirements are considered advantageous to some countries because they provide a way to collect nuclear export data needed for Additional Protocol declarations, as described in the previous paragraph. However, the procedure and the effort needed may not be proportionate to the scope. Moreover there is no requirement to report to IAEA the transfer of technology, which is instead included in Annex IV.

A joint meeting of the Dual Use Coordination Group and ENEF (European Nuclear Energy Forum) was also recently held on this topic. Regulatory framework's improvements are under consideration at Commission and Council.

EXP-WG will follow the evolution, but for the moment concluded that no specific research actions will be undertaken.

## 7. Internal Compliance programmes (ICP)

Exporters' compliance to export control requirements is a key element of the whole process. Without awareness and collaboration of exporters, who actively seek to fulfil the legal requirements, the only barrier to illegal exports would be enforcement at borders. Regulation 428/2009 attributes specific responsibilities to exporters, who have to make sure if their exported items appear on the applicable control list. Or if they are informed or aware of possible dual-use relevance, seek the opinion of authorities for a possible "catch-all" authorisation.

Different types of technology holders and suppliers need tailored compliance programmes. There is no universal model.
Compliance models depend on companies' decisions. There exist guidelines, but in the end suppliers decide on their own organisation and responsibilities.

From the regulatory point of view, some countries attribute a high values to ICP, whose existence may provide the company with facilitations towards the use of Open licenses (UK), or is even a mandatory requirement for Bulk General Licenses (Japan). In these countries, ICP may also be the object of audits.

A key element of an ICP is management's commitment, without which there is no real chance that the internal system may work.
Large companies have developed structured teams, mostly headed by a responsible in the legal department relying on distributed responsibilities and technical expertise in the various branches.
Mapping the portfolio of items produced and exported is mandatory, in order to improve the process.
This may be a complicated exercise for general dual-use items, and less for "especially designed or prepared" nuclear items.

Compliance needs to take into account all the applicable regulations, which may also be specific to the country of destination's sanctions. The company's internal review process should identify oddities in a procurement request and stop the process already at its start.

Large multi-national groups have branches all over the world. This means that also the movement of goods within the same company (e.g. for subsequent processing or assembly) is subject to controls. Inside the EU, this applies to the items listed in Regulation 428/2009 Annex IV.

Compliance has been identified as one of the key topic that EXP-WG plans to address by benchmarking existing ICPs and considering also models applicable to research and academic institutions.

## 8. Intangible Technology Transfers (ITT)

Trade may take different forms. Especially in the research environment, but also elsewhere, technology may be transferred also by intangible means, e.g. electronic mail, web-uploading / downloading; phone calls; intranet exchanges.

Reg. 428/2009 submits ITT to the same requirements as the ones imposed on tangible exports. Nevertheless a distinction is made for transfer of information by an expert travelling to another country, in which case we speak about "technical assistance", covered by a Council Joint Action [23].

Key conditions to assess are if the technology in object is "in the public domain" or "basic research", as well as possibly the minimum necessary for the installation of an equipment.
The first two are particularly challenging to judge upon, especially in the public research world. What is "basic" and what is in the public domain may not be clear at the start of a project or collaboration.

Technology for the "development, production or use" of an item, may remain controlled even if the item itself is not.

A particularly challenging aspect is posed by “cloud-computing”, which is nowadays a very modern approach to information technology. It is not clear yet if “cloud computing” is used for sensitive technology transfers and further investigation shall be conduct.

Three cases may be identified:

- users send numerical inputs to a code running in the cloud and receives back results. The inputs may refer to a controlled item (e.g. material properties) and hence possibly require an export license
- users develop their own code (e.g. modelling a controlled process) and transfer it to a server located in the cloud offering advantageous calculation speed and costs
- users use the cloud service also to store their possibly controlled information

Discussions are on-going in some relevant fora to identify the most suitable approaches to cloud computing from an export control perspective. Security aspects pertain to other domains.

ITT and cloud computing are modern challenges to compliance, and possibly the predominant issues for research.

EXP-WG plans to work on better understanding ITT in its various forms, identify existing guidelines best practices, issues and missing information, summarising the findings in technical sheets.

## 9. Conclusions

The paper recalled how export control of dual-use goods is an important barrier against the proliferation of weapons of mass destruction.

The new ESARDA working group on export controls was launched on October 15-16, 2012 to address research, technical, implementation and related issues.

Care was taken to avoid unnecessary duplications, overlapping and confusion with other existing initiatives, at least in the EU. EXP-WG has no political mandate, but compared to other groups dealing with the subject, presents the added value of including various types of contributors and stake-holders with different views and needs related to export control.

The first meetings were attended by nuclear regulators; suppliers and operators; R&D organisations; universities; SIPRI; IAEA and European Commission (JRC and ENER). There is an increasing interest which hopefully will be maintained and directed towards joint working activities and outputs in form of reports and non-binding guidelines (or technical sheets).

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# Investigation on the long-term stability of IRMM-1027 series of Large-Sized Dried (LSD) spikes

**Renáta Buják[a], Jeroen Bauwens[a], Rožle Jakopič[a], Maarten de Groote[b], Ludwig Cardon[b], Yetunde Aregbe[a]**

[a]Institute for Reference Materials and Measurements (JRC-IRMM)
Joint Research Centre, European Commission
Retieseweg 111, 2440 Geel, Belgium
[b]Faculty of Engineering and Architecture, Department of Materials Science & Engineering, Centre for Polymer and Materials Technologies (CPMT), Ghent University
Technologiepark 903, 9052 Zwijnaarde, Belgium
E-mail: renata.bujak@ec.europa.eu, yetunde.aregebe@ec.europa.eu

***Abstract:***

*The IRMM-1027 is a certified reference material for nuclear material control and accountancy. Each individual spike contains about 50 mg uranium (m($^{235}$U)/m(U) ~ 20%) and about 1.8 mg plutonium (m($^{239}$Pu)/m(Pu) ~ 98%) in dried form. They are used for the determination of U and Pu content of spent fuel solutions by isotope dilution mass spectrometry (IDMS) at the safeguards on-site laboratories. The dried uranyl and plutonium nitrates are embedded in an organic substance, which provides a stable layer at the bottom of the vial and preserves the integrity of the spike during storage and transport. Cellulose acetate butyrate (CAB) with 19 wt% butyryl content has been applied on the IRMM-1027 spikes for the last 10 years.*
*However, the cellulose matrix containing the spike material tends to flake off after about two years. In order to prolong the shelf life of the spikes, JRC-IRMM has been investigating CABs with higher butyryl contents – 35 wt% and 52 wt% –, to optimize the chemical stability and to achieve a longer lasting layer. Empirical tests have shown that the higher the butyryl content, the more stable and the more resistant the CAB is. Accordingly, future series of LSD spikes will be prepared with CAB containing 35 wt% butyryl, which has been proven to be intact for at least 4 years.*
*Furthermore, JRC-IRMM together with CPMT is investigating the chemical and mechanical properties of the three different CABs to gain scientific data on their stability and durability.*

**Keywords:** LSD spike; certified reference material; nuclear material accountancy; CAB; deterioration

## 1. Introduction

The IRMM-1027 series of LSD spikes are used for the determination of U and Pu content of spent fuel solutions by isotope dilution mass spectrometry (IDMS) at safeguards laboratories and nuclear facilities worldwide. The ratio of U:Pu=25:1 in the LSD spikes is designed according to the customers' specifications. The uncertainty that is introduced by the reference material is fit for safeguards purposes, and allows the laboratories to achieve IDMS measurement results with uncertainties below the respective target value of 0.28%, expressed as relative combined standard uncertainty [1].

CAB is a cellulose derivative with various amounts of hydroxyl, acetyl and butyryl groups. The mechanical and chemical properties of the CAB matrix are crucial for the stability of the organic layer and consequently influence the shelf-life of LSD spikes. CAB-19 (CAB with 19 wt% butyryl content) was applied for the first time in 2002 to replace the previously used tetrahydrofuran (THF) as protective organic matrix of the dried uranyl

and plutonium nitrates [2]. The advantages of CAB compared to THF are: easier preparation and handling in a glove-box, shorter drying period, and it readily dissolves in hot nitric acid. The chemical preparation of CAB-19 has been further improved [3] and it was demonstrated that the IRMM-1027 spikes have a shelf-life of two years. However, after this period the spikes start to crack, chip and flake off due to the deterioration of the CAB matrix. Therefore, JRC-IRMM has been testing besides CAB-19 also CAB-35 and CAB-52 with 35 wt% and 52 wt% butyryl content on the spikes to assess the long-term stability and robustness of the three different organic substances. The stability of the CAB and so the deterioration of the spike are affected by several factors such as: radiation [4]; acid residuals, humidity, and temperature [5]; physical shocks during transportation and mechanical stress in the thin layer [6]. Consequently, JRC-IRMM together with CPMT is investigating the chemical and mechanical properties of the pure CAB layers – without nuclear material – before and after ageing, irradiation and transport tests in order to find the best suitable matrix for future LSD spikes.

## 2. Long-term stability of LSD spikes

The processing of LSD spikes is demonstrated on the following flow chart (Figure 1).

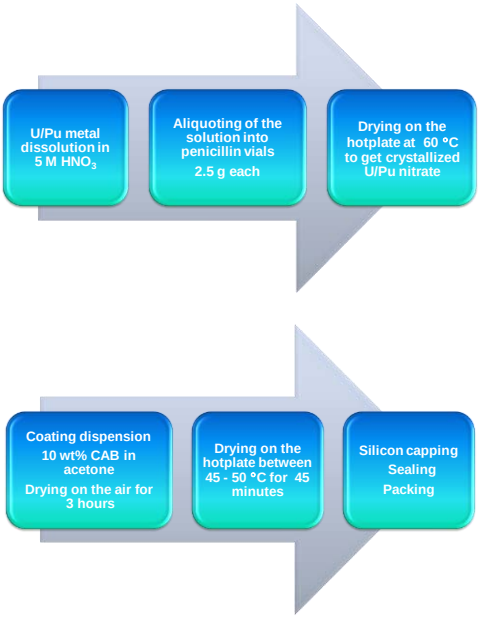


**Figure 1:** Preparation steps involved in LSD processing

The aliquoting of the nitrate solution is done by an automated system [7] while the drying and the CAB addition on the individual spikes are carried out manually in separate glove boxes. First, the nitrate solution containing the U and Pu is evaporated until complete dryness. As uranyl nitrate is hygroscopic and CAB is sensitive for moisture uptake, the humidity control is essential in the glove boxes. Then, the dried uranyl and plutonium nitrates are treated with a solution of CAB in acetone. Finally, the vials are capped and sealed in plastic bags.

At JRC-IRMM, empirical tests have been carried out to compare the three types of CAB. The preparation procedure for CAB-19, -35, and -52 was the same and the vials were stored and observed for two years. The outcome is shown on Figure 2. The vial containing the CAB-19 (No. 964) has a brownish colour and some cracks, while the spikes containing CAB-35 (No. 922) and CAB-52 (No. 586) are still intact. This is a good indication that CAB with higher butyryl content is more durable and provides a longer shelf-life for the spike material.


**Figure 2:** LSD spikes with CAB-19, CAB-35 and CAB-52 at the age of 2 years

These empirical tests are still on-going and visual observation is done regularly. The longest demonstrated shelf-life has been observed for test vials from the IRMM-1027M series that were covered with CAB-35 and have been intact already for four years. The CAB-52 was tested on the IRMM-1027L series and have not shown any flaking or cracking for two years. As a result, it was decided to use CAB-35 for the next series of LSD spikes IRMM-1027O. This would allow extending the shelf life on the IRMM-1027O certificate to three years. Furthermore, extensive studies have confirmed that the CAB-19 [2] and the CAB-35 [8] has no impact on the IDMS measurements and on the chemical treatment of spikes for the end users.

## 3. CAB: properties, stability and degradation behaviour

CAB is built up from repeating anhydroglucose units (monomers). Each monomer has three hydroxyl groups that can be esterified. The extent of esterification is expressed as mass percent (wt%) of butyryl group (see Figure 3). Its general features can be summarized as stiffness, moderate heat resistance, and moderate impact resistance. Some of the detrimental properties include a relatively narrow window between the melt flow temperature and the decomposition temperature due to the relative liability of the polysaccharide backbone at high temperatures [9].

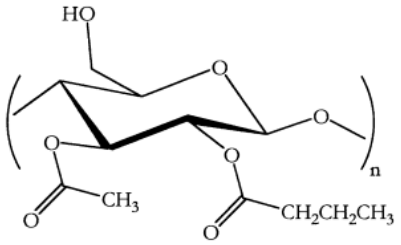


**Figure 3:** Cellulose acetate butyrate [9]

The chemical and mechanical stability of CAB are crucial for the long-term stability of the LSD spikes. There are several parameters which may influence the performance of the CAB layer. While it is possible to predict the sites of chemical attack in reactions such as hydrolysis or attack by acids, it is more difficult to predict the degradation pattern due to heat, light and radiation.

The drying temperature and the ambient humidity can be controlled during the spike preparation, however there are other parameters that cannot be fully controlled and could have a detrimental effect on the CAB matrix. These are the *radiation* coming from the spike material itself, natural *light*, *temperature* changes and *mechanical shocks* during transport and the combined effects of these.

The nuclides of Pu and U emit high energy alpha particles which are able to break bonds and create radicals or ions in the cellulose matrix. The calculated absorbed dose in the thin film is about $2\times10^5$ Gy which causes already moderate or severe effect on cellulose acetate [4]. The detrimental effect of radiation appeared on the samples as fissures, discoloration and releasing off gas as well as changes in the tensile strength [10]. However, there is still lack of exact data to draw detailed conclusion concerning the radiation effect on the cellulose acetate butyrate. Therefore, JRC-IRMM in co-operation with the Ghent University designed an irradiation experiment with X-rays simulating the long-term effect of radiation on CAB for one, two and three years. 100×100 µm samples will be casted of each CAB type and will be irradiated. The effect of radiation will be observed visually as well as the change in density, which is related to the degradation of the CAB, will be detected. The density will be monitored via the Compton scatter peak intensities.

The packing of radioactive material is carried out in two phases: the packing of the inner package and of the container. The IRMM-1027 LSD spikes are first sealed in plastic bags, then put in a plastic Type A container for radioactive materials, and finally they are transported in large sealed containers. For the transport of IRMM-1027 LSD spikes, currently the "EMMA package" is used according to relevant regulations [11]. Concerning the package material specifications and the fact that the transport of radioactive material does not take longer than one week, the units of IRMM-1027 are never exposed out of the temperature range of 4 to 40$^{o}$C. Although the spikes are well packed, possible shocks and temperature changes during different means of transport – plane and truck/car – could cause deterioration of the CAB. Therefore, further investigation underpinned by experimental data is needed. Preliminary test results of the different types of CABs on the chemical and mechanical properties carried out by CPMT confirm JRC-IRMM's empirical observations that the CAB-35 seems to be the best candidate to be applied on IRMM-1027 LSD spikes, thus it guarantees their longer shelf-life in the future.

Likewise, additional data are required about the ageing of the CAB layers due to heat, UV light and humidity in order to identify the most influencing parameters for the deterioration of the organic matrix.

## 4. Outlook

The shelf-life of LSD spikes covered with CAB-19 is about two years and there is a need from our end users to prolong it. JRC-IRMM has already optimized the chemical preparation of the CAB layer and has carried out empirical studies to test CAB with 35 and 52 wt% butyryl content as well. The studies have shown that the CAB with higher butyryl content is more

stable and durable; therefore it is more suitable for the future LSD spikes. Consequently, the IRMM-1027O series is covered with CAB-35.
Furthermore, additional studies will be carried out on how the ageing: light, heat, and humidity; radiation; and shocks during transport are affecting the durability of the three types of CAB.

## 5. Acknowledgement

The authors would like to thank to Prof. László Vincze from the Ghent University for his ideas in the irradiation tests.

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# Improvement of Precision of NRF-Based NDA using Monochromatic Gamma-Ray Beam for Nuclear Materials in Debris of Melted Fuel

**Toshiyuki Shizuma, Ryoichi Hajima, Takehito Hayakawa, Christopher Angell, Futoshi Minato, Kenya Suyama, and Michio Seya**

Japan Atomic Energy Agency
2-4 Shirakata-Shirane, Tokai, Ibaraki 319-1195, Japan
E-mail: shizuma.toshiyuki@jaea.go.jp

***Abstract:***

*Nondestructive assay (NDA) of actinide nuclei in spent nuclear fuel or melted fuel using a $\gamma$-ray beam is possible using a scattered method or a transmission method. These methods use nuclear resonance fluorescence (NRF) which is a process of nuclear excitation by photo-absorption and de-excitation by photo-emission. In the scattered method, system performance such as counting precision is affected by radiation background from the spent fuel and can be improved by using resonances at high energies where an optimum signal-to-noise ratio is obtained. In this talk, we will show calculated results on statistical uncertainties of the scattered method for various NRF cross sections in the range of 3 to 100 eV barn at excitation energies between 2 and 5 MeV. We will also mention the methodology of the transmission (self-interrogation) measurement and recent experimental results obtained using a quasi-monochromatic $\gamma$-ray beam.*

**Keywords:** non-destructive assay; nuclear resonance fluorescence; monochromatic gamma-rays

## 1. Introduction

Non-destructive assay (NDA) methods using a monochromatic $\gamma$-ray beam based on scattered [1] or transmission [2] techniques can be used to assay $^{239}$Pu in spent fuel. These methods use nuclear resonance fluorescence (NRF) and can be hopefully extended to assay melted fuel debris from the Fukushima accident. In the NRF scattered method, isotope-specific analysis is possible by directly detecting a characteristic NRF signal emitted from the isotope of interest. An efficient NRF measurement can be made using a monochromatic $\gamma$-ray beam which is generated by laser Compton scattering (LCS) [3,4]. This has been demonstrated in recent measurements for nuclear physics [5,6] and application studies [7,8]. The system performance such as statistical uncertainties of the NRF scattered NDA measurement would be affected by radiation background from spent fuel itself. Since the $\gamma$-ray spectrum of the radiation background has a characteristic shape with an exponential fall-off as a function of energy, NRF measurements at higher energies are preferable to obtain a better signal-to-noise ratio.

The transmission or self-interrogation method uses nuclear resonance absorption which removes resonant photons from the $\gamma$-ray beam. The selective isotope detection can be made by measuring the decrease of $\gamma$-ray beam intensity at resonance energies [9]. This method can be improved by integrating the signal over multiple resonances in the energy region of a quasi-monochromatic $\gamma$–ray beam, termed the Integral Resonance Transmission (IRT) method [2]. It uses all resonances within the energy window of the $\gamma$-ray beam, including weaker states that are not resolvable with current high-pure Ge detectors. The IRT method overcomes many limitations associated with directly measuring the scattered NRF $\gamma$-rays from the fuel [2].

Both the scattered and IRT methods need a monochromatic $\gamma$-ray beam. An intense monochromatic LCS $\gamma$-ray beam can be generated by combining two advanced technologies of an electron accelerator (energy recovery linac, ERL) and laser storing cavity [4]. The $\gamma$-ray beam energies are required to be

3.5 to 5 MeV for the scattered method and 2 to 3 MeV for the IRT method, corresponding to the electron energies of 450 to 500 MeV and 350 to 400 MeV, respectively.

In this paper, we will show calculated results for background radiation from the spent fuel obtained using a burn-up computer code and statistical uncertainties of the NRF scattered method for various integrated cross sections in the range of 3 to 100 eV barn at excitation energies of 2.3, 3.5, and 5 MeV. The NRF strength based on previous experimental data will be shown. We will also mention the methodology of the IRT measurement and recent experimental results obtained using a quasi-monochromatic $\gamma$-ray beam.

## 2. NRF scattered method

NRF is a process of resonant excitation of nuclear levels by absorption of photons and subsequent de-excitation to lower-lying levels by $\gamma$-ray emission. It occurs if the energy of the incident photon is identical to the resonance energy of the nucleus of interest. By measuring the intensity of scattered $\gamma$-rays, the amount of the specific isotope such as $^{239}$Pu can be quantified. The detailed methodology is presented in Ref. [10]. For the NRF scattered method, the activity from the fuel becomes significant background at energies below approximately 3 MeV where resonances are observed in actinide nuclei. Since the $\gamma$-ray strength of the radiation background exponentially decreases with increasing the $\gamma$-ray energy, NRF measurements at higher energies may provide a better signal-to-noise ratio. In the followings, we estimate the statistical uncertainties of the NRF scattered measurement at the energy region between 2 and 5 MeV by considering both of the NRF count rate and the background count rate from the spent fuel. We also examine the existing nuclear data for resonances above 3 MeV in actinide nuclei.

### 2.1. NRF count rate

The NRF yield can be obtained using an equation

$$Y_{NRF} = \phi \cdot I_s \cdot N_t$$

where $\phi$ is the intensity of the incident $\gamma$-ray beam, $I_s$ is the integrated cross section, and $N_t$ is the number of the target nucleus. When $\phi=10^6$ photons/s/eV, $I_s$=30 eV barn, and $N_t$ ($^{239}$Pu)=0.057g/cm$^2$ (effective thickness corresponding to Pu concentration of 1%), we obtain $Y_{NRF}$=3300 /s. Assuming an overall peak detection efficiency of $2.4\times10^{-4}$ for detection system consisting of 24 Ge detectors, the NRF count rate is calculated to be 0.8 counts per second (cps).

### 2.2. Background count rates

The detectors used for NDA of spent fuel may be subject to intense radiation background. Therefore, the analysis of the system performance should include effects from the radiation background. In the present study, the $\gamma$-ray radiation from the spent fuel was estimated using the ORIGEN2.2-UPJ computer code [11]. In case of a 10-year cooled spent fuel, photon intensity of radiation background was obtained to be $1\times10^{16}$ photons/s/MtU (Metric ton of uranium), evaluated for a typical PWR 17$\times$17 fuel assembly with initial $^{235}$U enrichment of 4.1 wt% under burn-up condition of 55 GWd/MtU. Figure 1 shows photon intensity from fission products and actinides as well as the total photon intensity. Fission products (actinides) mainly contribute to the photon intensity at energies below (above) 3 MeV. Assuming that the density of the spent fuel is 10.96 g/cm$^3$ and each detector points at a small part of the single fuel rod with a volume of 0.8 cm$^3$ which is determined by the size of the incident $\gamma$-ray beam, the background rate is estimated to be $1.1\times10^{11}$ photons/s. In this case, for a 50 kcps high-counting detector, the total detection efficiency including the intrinsic detector efficiency, the solid angle, and the absorption by radiation shields is required to be smaller than $4.4\times10^{-7}$. The photon intensities at E=2.3, 3.5, and 5 MeV are approximately $2.2\times10^{11}$, $1.1\times10^{9}$, and $2.2\times10^{7}$ photons/s/MeV/MtU, respectively. Therefore, the background radiation is estimated to be 1.9, $9.8\times10^{-3}$, and $1.9\times10^{-4}$ photons/s/eV at E=2.3, 3.5 and 5 MeV, respectively. Assuming a detector energy resolution of 0.2 %, the radiation background rates at the NRF peak positions are calculated to be 2.0,

$1.6\times10^{-2}$, and $4.6\times10^{-4}$ cps at E=2.3, 3.5 and 5 MeV, respectively. The values at E=3.5 and 5 MeV are considerably small in comparison with the NRF count rate mentioned above.

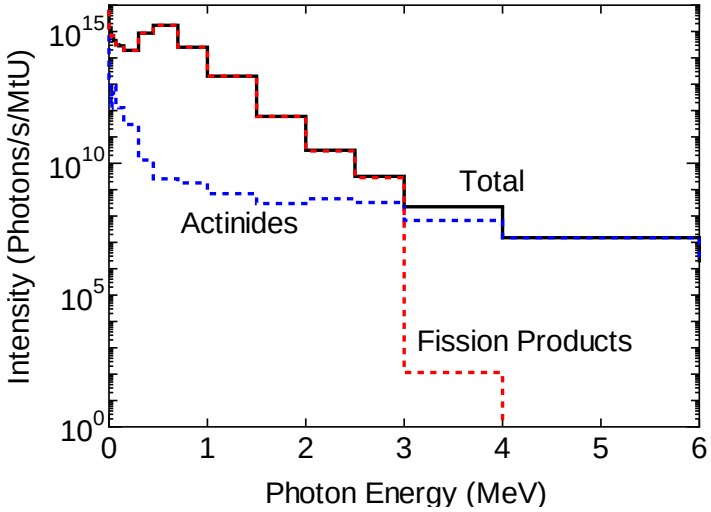


Figure 1; Photon intensities from a 10-year cooled PWR $17\times17$ spent fuel assembly with the initial $^{235}$U enrichment of 4.1 wt% under the burn-up condition of 55 GWd/MtU, evaluated using the ORIGEN2.2-UPJ computer code.

### 2.3. Statistical uncertainties

In a simple case, a statistical error is defined as

$$\frac{\sigma_S}{S} = \frac{\sqrt{\sigma_T^2 + \sigma_B^2}}{T - B},$$

where S, T, and B are the NRF count, the total count (T=S+B), and the radiation background count, respectively. The associated statistical deviations are denoted as $\sigma_S$, $\sigma_T$, and $\sigma_B$. Using the NRF and radiation background count rates discussed in Secs. 2.1 and 2.2, a 1% Pu content in spent fuel can be assayed with about 2% precision during 4000 seconds measurement time assuming NRF peaks located around 3.5 to 5 MeV with an integrated cross section larger than 30 eV barn.

### 2.4. Cross section data on actinides

In previous NRF experiments with linearly polarized photon beams, both magnetic dipole (M1) and electric dipole (E1) transitions have been observed at excitation energies between 2 and 4 MeV in $^{232}$Th [12] and $^{238}$U [13]. The scattering cross sections for the $1^-$ states at $E_\gamma$=4.002 MeV in $^{232}$Th and at $E_\gamma$=4.239 MeV in $^{238}$U have been measured to be 3.0 and 14 eV barn, respectively. In addition, scattering cross sections for $^{238}$U was measured by using monochromatic $\gamma$-rays emitted following neutron capture reactions [14]. The $\gamma$-rays with discrete energies at $E_\gamma$=4 - 6 MeV were used. In this measurement, strong NRF excitations in $^{238}$U were observed. The measured ground state decay widths are $\Gamma_0$=67, 24, and 15 meV at $E_\gamma$=5.090, 5.165, and 5.949 MeV, respectively, which correspond to the integrated cross sections of 30, 10, and 5 eV barn, respectively.

In actinide nuclei [12,13,15-18], resonance levels at $E_\gamma$=1.5 - 2.5 MeV were systematically observed with the scattering cross sections of tens of eV barn. These strengths are considered to carry a large fraction of the M1 scissors mode which is a common collective motion in deformed nuclei. On the other hand, at excitation energies higher than 3 MeV, electric dipole transitions are dominantly observed. Typical peak energies of E1 giant dipole resonances (GDR) in actinide nuclei are 10 to 12 MeV. Generally, in neutron-rich nuclei, E1 transitions, called the pygmy dipole resonance (PDR), are observed in the low-energy tail of the GDR. E1 transitions due to single particle excitation occur as

well. The observed low-energy E1 strengths in actinide nuclei are likely a combination of these E1 modes. From a viewpoint of nuclear structure, the existence of the afore-mentioned M1 and E1 collective motions are general in heavy nuclei. Therefore, the strong M1 and E1 resonances would exist at $E_x$=4 - 6 MeV in actinide nuclei such as $^{239}$Pu. The NRF measurement on $^{239}$Pu at $E_\gamma$=3 - 5 MeV is an important subject in the near future.

## 3. IRT method

A transmission or a self-interrogation method uses nuclear resonance absorption which removes resonant photons from the γ-beam beam. The selective absorption can be detected by measuring the scattered signal in a second target placed after the fuel. The second target is referred to as the witness target, which consists of the isotope of interest. This method is insensitive to the target geometry as well as the chemical composition, making it well suited for assaying the melted fuel at Fukushima as it will have a complex composition, and the size and shape is likely to be ill-defined [2]. The self-interrogation measurement can be improved by integrating the signal over multiple resonances in the energy region of a quasi-monochromatic γ–ray beam, termed the integral resonance transmission (IRT) method [2].

The IRT method requires a quasi-monochromatic γ-ray beam. The monochromatic nature of the beam eliminates the largest source of γ-ray background in the region of the resonant states of interest. The total integrated signal strength within the beam energy can be used. The average nature of the technique arises by using all resonances within the energy windows of the beam. Resonances with large widths will absorb more photons, while those with smaller width will absorb less. The formalism takes this into account by averaging over the depletion of all resonances. The method enhances the signal strength compared to using the conventional transmission method using only the strong single resonances. Using a monochromatic γ-ray beam, the total count rate in the detectors is reduced compared to using a bremsstrahlung beam, increasing the proportion of useable signal.

The schematic view of the IRT measurement is shown in Figure 2. The total flux of the γ-ray beam is first measured by a flux monitor. The beam traverses the melted fuel where resonance states in $^{239}$Pu selectively absorb γ-rays. The transmitted beam intersects with a scattering target consisting of $^{239}$Pu in a shielded measurement location where the emitted NRF γ-rays are measured using γ-ray detectors: either Ge detectors or scintillators. The total flux is measured again to normalize for attenuation from non-resonant processes, particularly atomic scattering. The NRF γ-ray intensity drops proportionally to the amount of $^{239}$P in the melted fuel. Two measurements will be done: first a measurement of the count rate without the melted fuel, and of the count rate with the melted fuel. The ratio of the two count rates is used to determine the total mass of $^{239}$P in the melted fuel (See Ref. [2] for more details on the IRT measurement).

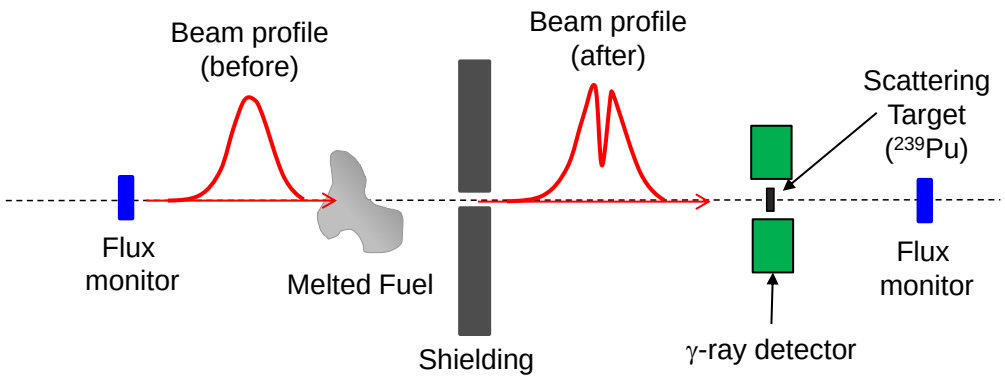


Figure 2; Schematic of integral resonance transmission (IRT) measurement. The $^{239}$Pu in the melted fuel resonantly removes photons. A decrease of the intensity of NRF signals from the scattering target is proportional to the amount of $^{239}$Pu in the melted fuel.

We recently carried out a measurement of the IRT method using a simulant target, $^{181}$Ta, whose nuclear resonant properties are similar to those of $^{239}$Pu. Measurements were made at 2.27 and 2.75 MeV, using a quasi-monochromatic γ-ray beam having an energy width of 4 %, at the High Intensity γ-ray Source (HIγS) facility, Duke University. The scattered radiation from the witness target was measured using HPGe and LaBr detectors. By placing either a Pb or Ta upstream of the scattering target, the resonant absorption due to states in $^{181}$Ta can be clearly observed as shown in Figure 3.

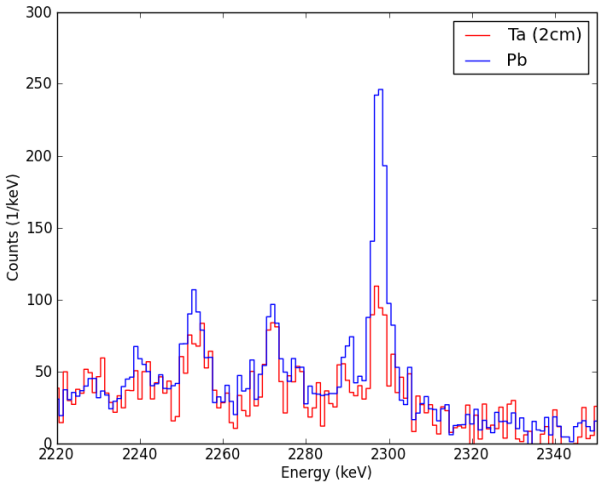


Figure 3; The scattering spectrum using a Ta target for either a Ta or Pb absorber placed upstream. The spectra have been normalized for same flux on target, demonstrating resonant absorption of photons by the Ta target.

## 5. Conclusion

Count rates of both the NRF and radiation background from the spent fuel were estimated using the burn-up code ORIGEN2.2-UPJ for a typical PWR fuel assembly. It is possible to assay 1% Pu content in the spent fuel with about 2 % statistical error during 4000 second measurement time for NRF peaks at E=3.5～5 MeV with an integrated cross section of 30 eV barn by using a monochromatic γ-ray beam with an intensity of $10^6$ photons/s/eV. It is concluded that the statistical uncertainties for the NRF scattered method can be improved by using resonances at high energies (above 3 MeV) where the optimum signal-to-noise ratio can be obtained. The previous experimental data on actinide nuclei show resonances with the integrated cross sections of several tens of eV barn, indicating that strong resonances would exist above 3 MeV in $^{239}$Pu. The NRF measurement on $^{239}$Pu at $E_\gamma$=3 - 5 MeV is an important subject in the near future.

The self-interrogation measurement can be improved by integrating the signal over multiple resonances in the energy region of a quasi-monochromatic γ–ray beam, termed the Integral Resonance Transmission (IRT) method. To demonstrate and verify the IRT method, we measured the IRT signature from a simulant material, $^{181}$Ta, whose nuclear resonant properties are similar to those of $^{239}$Pu. Measurements were made at 2.27 and 2.75 MeV, using metallic Ta targets, for both the absorption and scattering targets, using a quasi-monochromatic γ-ray beam. The results confirm the proposed IRT method.

## Acknowledgements

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# Measurement technique of plutonium using nuclear resonance fluorescence with laser Compton scattering gamma-rays

**Takehito Hayakawa, Toshiyuki Shizuma, Christopher Angell, Ryoichi Hajima, Mamoru Fujiwawa, Michio Seya**

Japan Atomic Energy Agency
Shirakata-shirane 2-4, Tokai-mura, Ibaraki-ken 319-1195 Japan
E-mail:hayakawa.takehito@jaea.go.jp

***Abstract:***

*We have proposed a new non-destructive assay system for $^{239}$Pu and other actinides in spent nuclear fuel assembly in a water pool and melted fuels. Nuclear fuel materials are detected using nuclear resonance fluorescence (NRF) with laser Compton scattering (LCS) gamma-rays, which are generated by collision of high energy electrons and laser photons. This method can identify each nuclide in any form and with any chemical property. We have demonstrated this method using available LCS gamma-ray beam. We have also designed an extremely high-flux LCS gamma-ray source based on a new generation electron accelerator of the energy recovery linac. A test facility based on an ERL has been under construction and the first generation of the ERL-LCS gamma-rays is scheduled in 2014 JFY (Japanese Fiscal Year).*

**Keywords:** non-destructive assay; Pu-239; gamma-ray

## 1. Introduction

Non-destructive assay (NDA) of plutonium and other actinides in various types of materials is one of key technologies for nuclear security and safeguards of future nuclear fuel cycle. Bertozzi et al. have proposed nuclear resonance fluorescence (NRF) using bremsstrahlung gamma-rays for nuclear security and other industrial applications [1]. However, the bremsstrahlung gamma-rays generally produce large background, in particular, in the low-energy region; signal-to-noise ratio decreases. The key technology for precise measurements of fissionable isotopes inside shields consisting of various elements is the monochromatic gamma-ray source.

Pruet et al. [3] has proposed a novel non-destructive detection method of $^{235}$U hidden in a cargo transporter by using laser Compton-scattering (LCS) gamma-rays. This system is expected to be used at the gates of important facilities, for example airport, seaport, and other important building areas. We have also proposed an extremely brightness LCS gamma-ray system based on energy recovery linac. The LCS gamma-ray beam is generated by inversed Compton scattering of laser photons with high energy electrons (see Fig. 2). The LCS gamma-ray sources can generate the energy tuneable quasi-monochromatic gamma-rays. NRF with LCS gamma-rays has the following advantages. First, one can measure the isotopic composition of materials through heavy shields like several centimeter-thickness metal plates and several ten centimeter-thickness of water since the energies of LCS gamma-rays are of the order of several MeV. Second, this method is, in principle, applicable to the non-destructive detection of unstable isotopes as well as stable isotopes of all elements except hydrogen. Third, one can detect NRF-scattered gamma-rays with a high signal-to-noise (S/N) ratio since the LCS gamma-ray source can produce a quasimonochromatic photon beam. Forth, the measured quantity of a nucleus of interest is insensitive to the chemical composition and form of an object. These advantages suggest this method is effective to measure $^{239}$Pu in complex materials, in particular, the melted fuel. The severe loss of coolant accidents of the Fukushima-Daiichi NPP caused by the earthquake and the Tsunami on March 11 have produced melted fuel in the reactor cores of Units 1 – 3. The fuels have been melted down and mixed with many types of objects at the inside of the reactors. We would like to stress that this method can measure 7 nuclides of $^{235}$U, $^{238}$U, $^{238}$Pu, $^{239}$Pu, $^{240}$Pu, $^{241}$Pu, and $^{242}$Pu with a high brightness LCS gamma-ray source. In the framework of the safeguards, it is required to measure the quantities of all of these 7 nuclides.

LCS gamma-ray sources in the energy range of several MeV have been developed at the National Institute of Advanced Industrial Science and Technology [4], the Duke Free Electron Laser Laboratory at Duke University [5], and at NewSUBARU in SPring-8 [6]. These facilities have been used for study of fundamental science such as nuclear physics and applications. We have demonstrated to detect isotopes concealed inside a heavy shield with an available LCS gamma-ray source for the first time. We have measured one-dimensional mapping of an isotope, $^{208}$Pb, shielded with an iron box with a thickness of 15 mm by NRF with the LCS gamma-rays [8]. We have also measured two isotopes of $^{12}$C and $^{14}$N in a chemical compound, melamine ($C_3H_6N_6$), through a 15-mm thick iron plate and a 4-mm thick lead plate [9]. This method is effective to measure the shape of hidden materials; we measured the shape of the triangular prism made of natural lead inside the 15-mm thick iron box. An advantage of this method is that we can identify isotopes as well as elements. We have measured distribution of two different isotopes of lead, $^{206}$Pb and $^{208}$Pb, in the iron box [11]. These results with the LCS gamma-ray beam show clearly this method is a power tool.

The key technology for actual system is generations of high brightness LCS gamma-ray beam. As such a system, we have proposed a high-flux gamma-ray facility utilizing a 350-MeV energy-recovery linac (ERL) equipped with a superconducting accelerator [7]. In this paper, we present the design of this system and status of the development of a test facility at KEK.

## 2. Measurement using nuclear resonance fluorescence with laser Compton scattering gamma-rays

Figure 1 shows a schematic view of NRF with the monochromatic gamma-ray beam. If the energy of the incident gamma-ray is identical with the energy of an excited state populated directly from the ground state of the nucleus of interest, the incident gamma-ray is effectively absorbed in the nucleus and subsequently the nucleus re-emit the gamma-ray. The energies of the states excited by NRF are inherent in the atomic number and mass of the nucleus of interest as shown in Fig. 1. By measuring the energies of the NRF gamma-rays, we can analyse nuclear species. The number of each isotope can be evaluated by the number of an NRF peak in the measured energy spectrum.

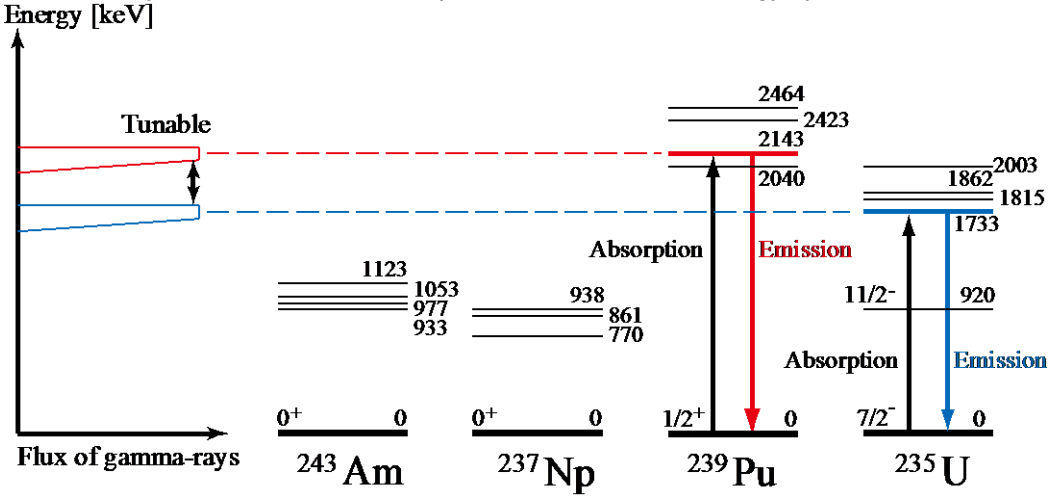


Figure 1 Schematic view of nondestructive assay using nuclear resonance fluorescence with monochromatic gamma-rays

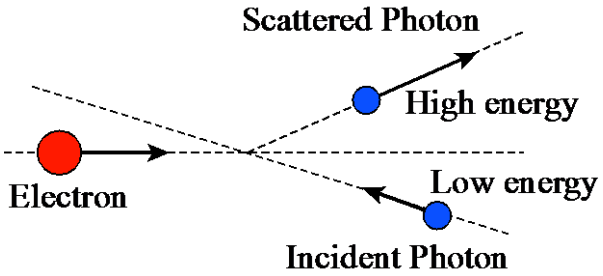

Figure 2 Schematic view of laser Compton scattering gamma-rays

## 3. New laser Compton scattering gamma-ray source based on energy recovery linac

Gamma-rays are produced via inverse Compton scattering of laser photons with relativistic electrons. In order to increase the flux of a laser Compton gamma-ray source, the density of laser photons and electrons at the collision point must be increased. In addition the collision area should be decreased. The collision density, however, has a practical limitation for several reasons: i) laser power and electron beam current are practically limited, ii) the laser spot size is constrained by the geometry of laser beam optics, iii) the electron beam size is determined by the beam emittance.

In order to increase the flux of LCS gamma-ray beams by several orders of magnitudes than that of the available LCS gamma-ray facilities, we have proposed a extremely high-flux LCS gamma-ray facility based on energy-recovery linac (ERL) equipped with a superconducting accelerator and a laser enhanced cavity [7]. The ERL has a great impact on modern accelerator applications. The ERL accelerators can generate a high-quality electron beam with a high-intensity average-current. An ERL is composed of an injector with an electron gun, a superconducting linac, an energy recovery loop, and a beam dump. An electron beam from an injector is accelerated by time-varying radio-frequency (RF) field stored in a superconducting linear accelerator and subsequently is transported to a recirculation loop. Each electron bunch in the recirculation loop is used for applications such as generation of gamma-rays. After the recirculation, the electron beam enters again the superconducting cavity for the deceleration RF phase. The re-circulated electron is decelerated and feeds back the energy to the superconducting RF cavity. This recycled RF energy is again used to accelerate subsequent electrons. Electron bunches after deceleration go to a beam dump. The advantage of the ERL comes from the fact that each electron bunch is used only once for application such as a gamma-ray generation and thus the ERL is free from degradation of electron beam emittance.

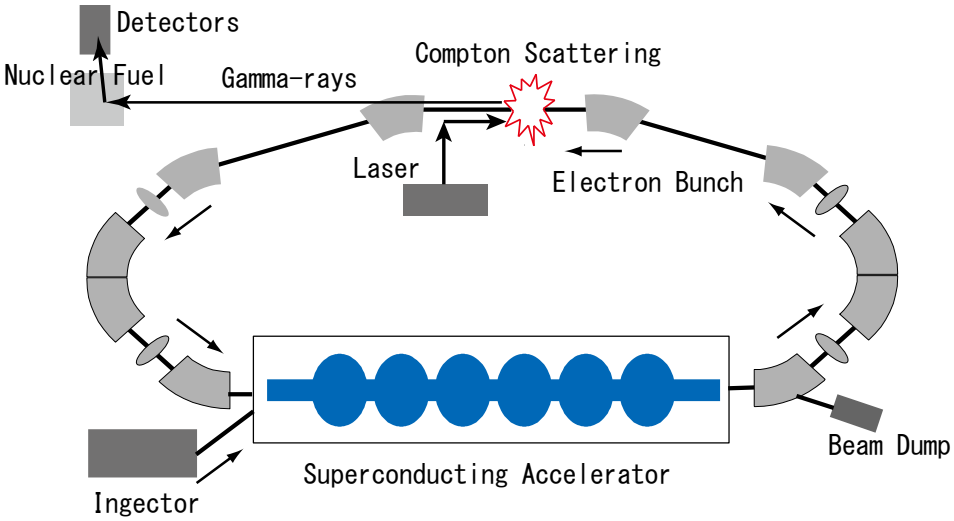

Figure 3 Schematic view of the next generation of LCS gamma-ray beam source based on the energy recovery linac.

Since the ERL with electron energy in order of GeV can generate extremely high-brightness and short pulse light, the ERL is a candidate for the next generation light source facility after the third generation synchrotron and the x-ray free electron laser facility. In fact, there are several ERL projects in the world. One of the projects is the Compact ERL (cERL) project at the high energy accelerator research

organization (KEK) in Japan [14]. This cERL is a test facility to demonstrate the generation of ultra-low emittance electron beams for a future synchrotron light source based on a 3-GeV ERL, which has been proposed at KEK. The cERL facility is used also for demonstration of the generation of the LCS gamma-ray beam. The cERL comprises a 500-kV DC photocathode electron gun, a low-energy beam transport line, a 1.3-GHz buncher cavity, three 2-cell superconducting cavities, and laser enhanced cavity (see Fig. 4). A 500-kV gun, which was developed at the Japan Atomic Energy Agency (JAEA), is installed. The beam energy of the injector is 5 MeV and the electron is accelerated up to 35 MeV by the superconducting cavity. The laser cavity consists of a mode-lock laser and mirrors to accumulate the laser power from the initial laser power by 1000 inside the cavity. The LCS gamma-rays are generated by the collision between the ultra-low emittance electron bunches and laser photons inside the laser cavity. The generated gamma-ray beam is introduced to an experimental room. This system is under the construction (see Fig. 5) and the demonstration of basic technology of LCS gamma-ray beam generation is scheduled in 2014 JFY at this moment.

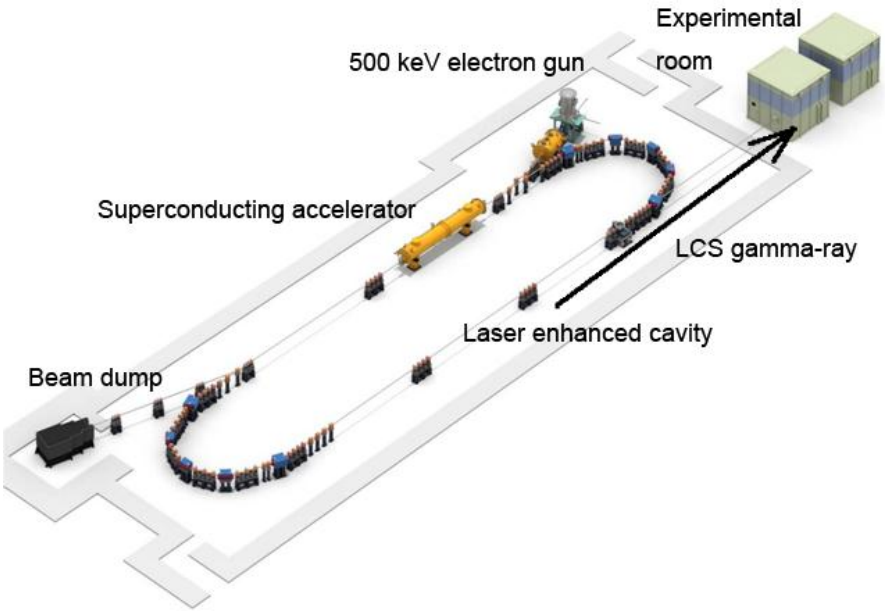


Figure 4 Schematic view of the ERL test facility developed at KEK.

Figure 5 Pictures of radiation shields for accelerator (left) and installed electron gun (right)

We would like to point out that the emittance of electron beam is important for the energy width of the generated LCS gamma-ray beam as well as the flux. The energy of each scattered photon depends on the scattered angle, and a quasi-monochromatic gamma-ray can be obtained by putting downstream a collimator to restrict the divergence of the gamma-ray beam. Under the same geometry with the same collimator, the energy width of the LCS gamma-ray beam decreases with decreasing the emittance of the electrons. A typical energy resolution of available LCS gamma-ray beams is about

dE/E ~ 3-10%, whereas a sharp energy resolution of dE/E ~ 0.1% can be generated by the ERL based LCS gamma-ray facility.

## 4. Development of simulation code

We performed a Monte Carlo simulation to study a proposed non-destructive assay system. Interactions of the generated gamma-rays with an object to be measured are studied by using the code GEANT4, which is a package of toolkits for the simulation of the passage of particles through matter. Although the toolkit for the GEANT4 is widely used in various science fields, the library to calculate the NRF process is not supported in the original GEANT4 package. We have, therefore, modified the code GEANT4 to calculate the NRF process in cascade interactions of photons and atomic nuclei. There are three libraries for calculation of fission, inelastic scattering, and elastic scattering for hadrons. We have added three libraries, MyGammaElasticProcess, MyGammaCrossSectionDataStore, and MyGammaElasticCrossSectionData. These libraries are extensions to calculate NRF, where we should input cross section (energy width) for NRF and excitation energies as nuclear data. We traced a track of each photon and the change of its energy taking into account atomic processes and nuclear reactions. To reproduce the measured NRF spectra, we implement some physical processes as the Doppler broadening of scattered gamma-ray, the energy loss caused by the recoil. We also consider gamma-ray angular distributions from nuclei. The angular distribution depends on the linear polarization of the incident LCS gamma-ray, the initial spin and parity of the ground state of the target nuclei and excitation mode by the incident photons.

Another simulation code MCNPX has been used widely in nuclear engineering and has been extended independently to calculate NRF. We JAEA and LBNL (LANL/ORNL) have a collaboration of improving the NRF simulation codes of both side using a same experimental data obtained at HIGS of Duke University.

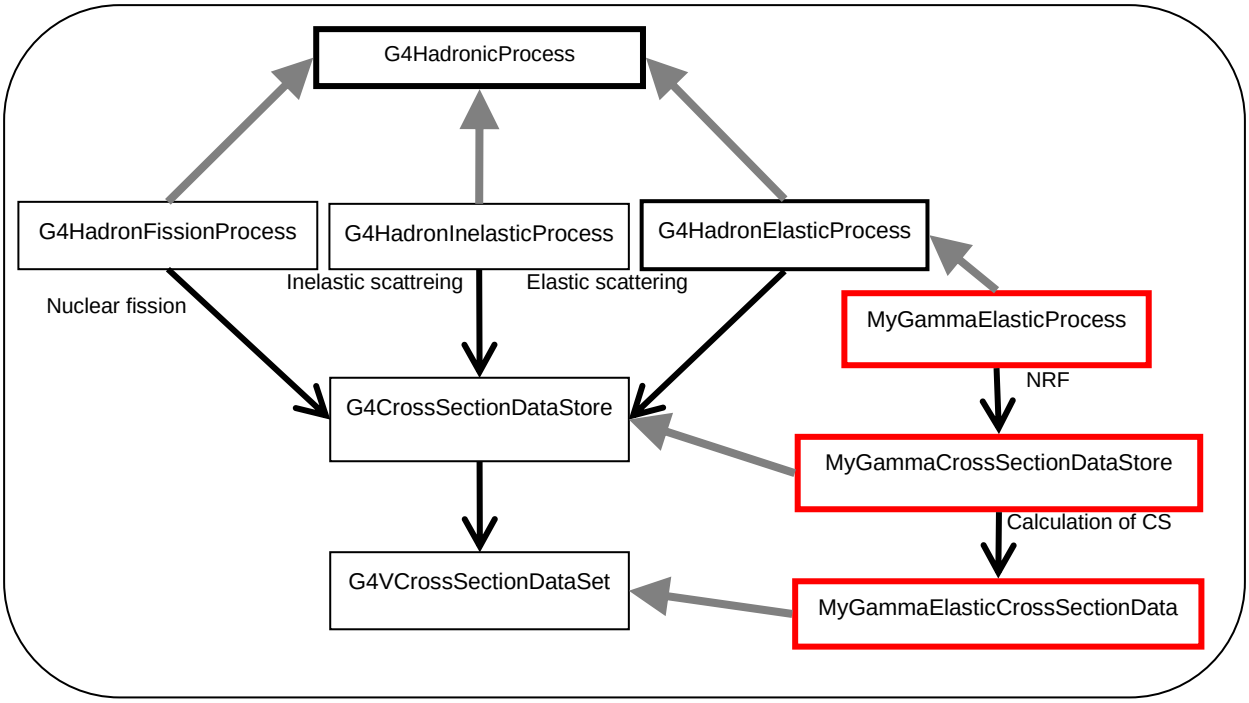


Figure 6 Block diagram of extended GEANT4

## 5. Proposal of the NDA systems using laser Compton scattering gamma-rays

### 5.1 NDA system of Pu in fuel assemblies constructed with nuclear power plants

We have proposed a non-destructive assay system for $^{239}$Pu and other actinides in spent nuclear fuel assembly in a water pool [12]. Figure 7 shows a schematic view of detection system. The spent fuel

assembly is kept in a water pool, whereas a gamma-ray detection system is located outside of the water pool. The spent fuel contains many radioactive isotopes and becomes feverish. Thus, the spent fuel is stored in a water pool for cooling. Neutrons from spontaneous fission of actinides in the nuclear fuel may be critical problem for detection of gamma-rays but the neutrons are absorbed by water. The gamma-rays emitted from the spent fuel are measured with high-purity germanium (HPGe) detectors in outside of the water pool. The advantage of the HPGe detectors is a good energy resolution, which is typically smaller than 0.2% (full width at half maximum (FWHM)). For detection of NRF gamma-rays, a multi gamma-ray detector array of the HPGe detectors is used, which has been widely used for the study of nuclear physics. This type of detector system typically consists of 20-120 HPGe detectors. In the present calculation, the detector array consists of twenty-four HPGe detectors in the present system (see Fig. 7). A lead collimator with a thickness of 15-20 cm and a lead absorber with a thickness of 5-8 cm are located in front of each HPGe detector. The collimators are used to limit their solid angles. The absorbers are used to reduce the counting rate of the detectors. The gamma-rays with energies lower than 1 MeV are effectively absorbed by these thick absorbers. A simulation calculation shows that we can detect 1% fraction $^{239}$Pu in all the fuel rods with statistical error lower than 2% using the high flux LCS gamma-ray source and the measurement time of 4000 s.

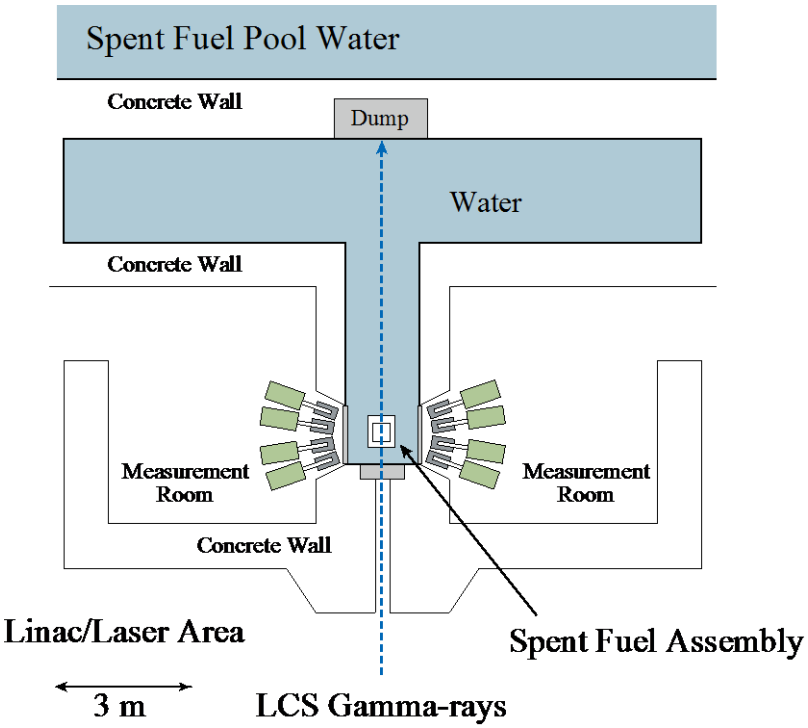

Figure 7 Schematic view of an NDA system of Pu in spent fuel assemblies.

## 5.2 ERL based hybrid K-edge/XRF densitometry system

We have proposed a new system ERL based hybrid K-edge/XRF densitometry (HKED) system with monochromatic (energy adjustable) X-rays of 120-150 keV [13]. The HKED system has been used for measurements of density of several key elements in solutions in nuclear fuel processing plants and re-processing plants of spent nuclear fuels. The HKED system combines two assay techniques of the K-edge densitometry (KED) and the X-ray fluorescence (XRF) densitometry. The KED technique measures the transmission of a photon beam through a sample to determine the concentrations of major ingredients in solutions. On the other hand, the XRF technique measures characteristic X-rays from the atoms which are excited by incident photons with energies above the K absorption edge. Bremsstrahlung radiations from an X-ray tube are used as an incident photon source in the available system. The bremsstrahlung radiation is characteristic of strong photon intensity but its broad energy distribution causes large background due to inelastic scattering. In the photoelectric process, photons with energies above the K-edge are only absorbed, and subsequently characteristic X-rays are emitted. If the energy spreading of a photon beam is narrowed and the beam energy is set to just above the K absorption edge, the sensitivities for the XRF detection of the HKED system will be improved

dramatically by increasing the peak-to-background ratio. Therefore, we have proposed an advanced HKED method with a monochromatic photon beam generated by an ERL based LCS system.

## 5.3 NDA systems of nuclear material in melted fuel debris

The shape and chemical property of the melted fuel in the Fukushima plants have not been known because in the present time there is no method to measure the inside of the reactors. It is considered that there are several types of materials in the reactors. In the bottom of the pressure vessel in the reactor there is a large block mixed with many objects such as spent fuel rods, frames of fuel assemblies, reactor control rods, and covers of fuel rods. Several strata of removed melted fuels can be categorized into particle-like debris, small rock-like debris, cylindrical debris, and slab type debris. The size of the particle-like debris may be several millimetres. The object of the second size class, small rock-like debris, would be materials with a size of several centimetres or a few tens centimetres. The third class object is debris larger than 1 m, and we should make a small size of samples like cylindrical debris from the large debris to manipulate samples in the NDA system. For another example, the debris with the size of 5-10 cm can be stored to a case.

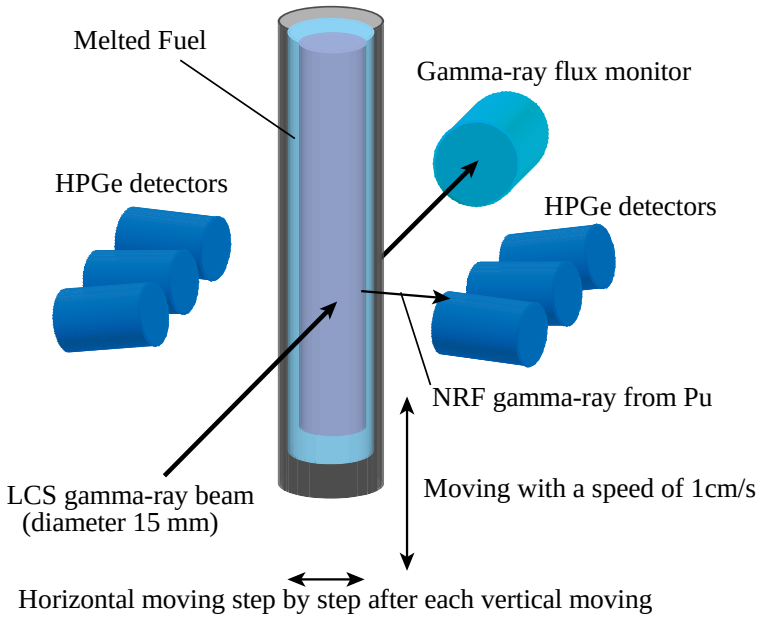


Figure 8 Schematic view of an NDA system for melted fuel debris with measuring of scattered NRF gamma-rays.

For examples, we consider two procedures to measure the fraction of $^{239}$Pu in the melted fuel. The first method is to measure directly scattered gamma-rays from the sample with a LCS gamma-ray irradiation and evaluate the quantity of $^{239}$Pu from the yield of a $^{239}$Pu NRF peak in the measured gamma-ray energy spectrum. Another method is the self-interrogation method (notch method or witness method) to measure the absorption of photons due to the nucleus involved inside of the sample.

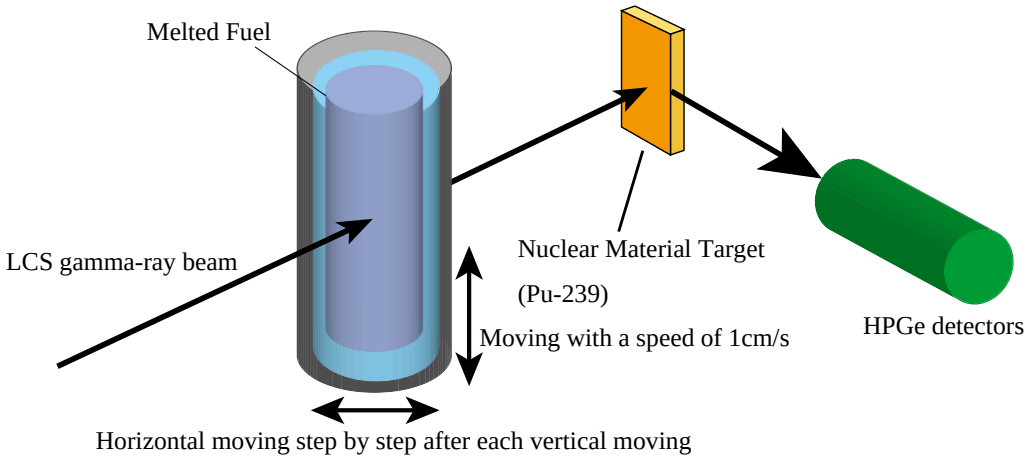


Figure 9 Schematic view an NDA system for melted fuel debris with self-interrogation method

Figure 8 shows a schematic view of the NDA system with measuring the scattered NRF gamma-rays from the sample. The melted fuel is kept in water inside of the case. Gamma-ray detectors are located at the outside of the case. The melted fuel contains a large amount of radioactive isotopes and becomes feverish. Thus, the melted fuel is stored in water for cooling. The gamma-rays emitted from the melted fuel are measured with high-purity germanium (HPGe) detectors. An advantage of the HPGe detector is a good energy resolution, which is typically lower than 0.2% (full width at half maximum (FWHM)). For the detection of NRF gamma-rays, a multi gamma-ray detector array of many HPGe detectors is used. The detector system consisting of 20-120 HPGe detectors has been widely used for the study of nuclear physics. In the present calculation, the detector array consists of twenty-four HPGe detectors. Eight HPGe detectors are located in a plane, and each detector measures only gamma-rays emitted from each small area with a collimator to limit the solid angle. With the multi detectors, we can measure the $^{239}$Pu distribution along the axis of the incident beam.

The melted fuel is vertically moved with a slow speed, for example, with 1 cm per 1 second, to measure all the position. After a sequential measurement as a function of the vertical position of the fuel sample, the sample is horizontally moved for the diameter of the incident LCS gamma-ray beam. In this way we measure the three-dimensional image of 239Pu in the sample.

The second method is the self-interrogation method (see Fig. 9). This method could be extended to cover small resonances vicinity of large resonance to make absorption cross section larger than a single large resonance, which we call Integral Resonance Transmission (IRT) method. The sample is irradiated by the LCS gamma-ray beam with energy identical with the energy of an excited state of $^{239}$Pu. In general the resonance width of the excited state is in order of meV or tens meV, whereas the energy spread of the incident LCS gamma-ray beam is much larger than the resonance width even if the energy spread dE/E is only 1%. If $^{239}$Pu exists at the inside of a sample, the incident gamma-ray whose energy is same as the excitation energy is effectively absorbed. The energy spectrum of the penetrated gamma-ray beam has a deep valley at the top part of the peak and the depth of the valley is in general proportional to the quantity of $^{239}$Pu at the inside of the sample. The penetrated gamma-rays make NRF on $^{239}$Pu in the target (witness plate). With measuring the scattered gamma-rays from the witness plate, we can measure the yield of a $^{239}$Pu NRF peak. The yield is proportional to the intensity of the penetrated gamma-ray beam at the $^{239}$Pu excitation energy, which is equal to the initial gamma-ray intensity minus the absorption on the sample corresponding to the depth of the valley (see Fig. 9). Since we know the initial intensity and the thickness of the witness plate, we can evaluate the quantity of the $^{239}$Pu at the inside of the sample involved. This method has an advantage that we can measure photons with a low counting rate since the background gamma-rays from the melted sample are shielded.

## 6. Summary

We have proposed a new non-destructive assay system for $^{239}$Pu and other actinides in spent nuclear fuel assembly and melted fuels in water pools. Nuclear fuel materials are detected using nuclear resonance fluorescence (NRF) with laser Compton scattering (LCS) gamma-rays, which are

generated by collision of high energy electrons and laser photons. We have designed an extremely high-flux LCS gamma-ray source based on a new generation electron accelerator: the energy recovery linac (ERL). An ERL has been constructed at KEK in Japan. The generation of the ERL based LCS gamma-ray beam is scheduled for 2014 JFY.


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# Proposal of Neutron Resonance Densitometry for Particle Like Debris of Melted Fuel using NRTA and NRCA

**H. Harada[1], F. Kitatani[1], M. Koizumi[1], H. Tsuchiya[1], J. Takamine[1], M. Kureta[1], H. Iimura[1], M. Seya[1], B. Becker[2], S. Kopecky[2], P. Schillebeeckx[2]**

[1]Japan Atomic Energy Agency (JAEA)
Tokai-mura, Naka-gun, Ibaraki 319-1195, Japan
E-mail: harada.hideo@jaea.go.jp

[2]EC-JRC-IRMM
Unit D.4. Standards for Nuclear Safety, Security and Safeguards
Retieseweg 111, B-2440 Geel, Belgium
E-mail: peter.schillebeeckx@ec.europa.eu

***Abstract:***

*Neutron resonance densitometry (NRD) has been proposed to quantify nuclear materials in particle-like debris of melted fuel formed in a severe accident of nuclear reactors such as the Fukushima Daiichi nuclear power plants. NRD is a method that combines NRTA (neutron resonance transmission analysis) and NRCA (neutron resonance capture analysis) using a pulsed neutron generator and the TOF (time of flight) technique. NRTA is used to quantify the amount of Pu and U isotopes. NRCA is used to identify matrix materials, such as B and Fe, which are present in the melted fuel. A special gamma-ray spectrometer has been designed to apply NRCA in the presence of highly radioactive materials. The applicability of the NRD method has been studied using Monte Carlo simulations and neutron TOF experiments at the GELINA facility of the EC-JRC-IRMM. We conclude that NRD has a potential to determine the quantities of Pu and U isotopes in particle-like debris of melted fuel with counting statistics uncertainties less than 1%, even in the presence of 2.5 w% $^{nat}$B and 9 w% $^{56}$Fe.*

**Keywords:** neutron resonance densitometry; neutron time-of-flight; melted fuel; debris; NRTA; NRCA

## 1. Introduction

An innovative non-destructive analysis technique, referred to as neutron resonance densitometry (NRD), is presented in this paper. This method was proposed to quantify nuclear materials in particle-like debris of melted fuel formed in a severe accident of nuclear reactors such as the Fukushima Daiichi nuclear power plants. NRD is a method combining NRTA (neutron resonance transmission analysis) and NRCA (neutron resonance capture analysis).

The NRTA technique was applied in the past to quantify nuclear materials in irradiated nuclear fuel pellets by Priesmeyer and Harz [1] using a fast chopper time-of-flight (TOF) spectrometer and by Behrens *et al* [2] using a pulsed neutron TOF spectrometer. Recently, Sterbentz and Chichester have proposed NRTA as a non-destructive assay technique for the next generation safeguards initiative's plutonium assay challenge [3].

Behrens *et al* [2] have analysed neutron transmission spectra covering a neutron energy range between 1 and 40 eV. The abundances of $^{234\text{-}236}$U and $^{238\text{-}242}$Pu isotopes relative to $^{238}$U were deduced with uncertainties ranging from better than 1 % for $^{235}$U to 20 % for $^{238}$Pu. In the analysis, they have taken into account 16 nuclides (11 actinides and 5 fission products) with a resonance structured cross section and oxygen as matrix of the sample and the steel end cap. Both, the oxygen and steel, have a slowly varying cross section without resonance in this energy region.

On the contrary, a detailed elemental and isotopic composition of melted debris is not known. Very likely, the samples will contain water, boron, structural materials, and concrete materials. However, no information about their relative amounts is available. It is important to identify which components are included in the melted debris in advance of the NRTA measurement to optimize the analysis of the data. NRCA is used for identifying possible contaminants, such as $^{10}$B and $^{56}$Fe. A well-type gamma ray spectrometer made of a $LaBr_3$ scintillation detector was proposed and designed. This detector is optimized to discriminate against the intense $\gamma$ rays emitted from the debris of melted fuels due to the decay of $^{134,137}$Cs.

Both NRTA and NRCA have been developed and applied at the GELINA facility of the EC-JRC-IRMM as non-destructive analysis methods [4]. The Japan Atomic Energy Agency (JAEA) and EC-JRC-IRMM have started a collaboration on the feasibility study and the development of NRD since 2012 JFY.

In this paper, the principles of NRD are given in chapter 2; the experimental study in chapter 3; the design of a practical facility in chapter 4; an evaluation of uncertainties in chapter 5. A part of the concept of the NRD is described in refs. [5, 6].

## 2. Principles

The total neutron cross section is one of the most accurately determined nuclear data, and neutron resonance parameters of major actinides are well known in the resolved resonance region. Therefore, NRTA is expected to be one of the most accurate non-destructive analysis techniques. Fundamental principles of NRTA and NRCA have been well described in ref. [4]. In the proposed NRD, resolved resonances are used to identify and quantify the Pu and U isotopes, whose resonance energies are less than 50 eV. To resolve the resonances in this energy region, a short neutron flight path length of about 5 m can be applied as discussed in ref. [7].

For water, boron, structural materials and materials used in concrete, there are no resonances present in the energy region below 50 eV. However, most of the light and medium elements present in these materials emit discrete prompt $\gamma$ rays with significant intensities via neutron capture reactions. By detecting these discrete prompt $\gamma$ rays, these elements can be identified. In **Table 1**, energies of prominent prompt $\gamma$ rays are summarized for some elements expected to be included in the debris.

Except $^{10}$B, all of the listed isotopes emit $\gamma$ rays of energies much larger than the 661-keV $\gamma$ ray following the decay of $^{137}$Cs. Therefore, measurements of these high energy $\gamma$ rays do not suffer from a Compton background due to 661-keV $\gamma$ rays.

| Nucleus | Reaction | Energy of Prominent Prompt γ ray | Energy of 1st Neutron Resonance |
|---|---|---|---|
| $^{1}$H | $^{1}$H (n, $\gamma$) $^{2}$H | 2223 keV | — |
| $^{10}$B | $^{10}$B (n, $\alpha\gamma$) $^{7}$Li | 478 keV | 170 keV |
| $^{28}$Si | $^{28}$Si (n, $\gamma$) $^{29}$Si | 3539, 4934 keV | 31.7 keV |
| $^{56}$Fe | $^{56}$Fe (n, $\gamma$) $^{57}$Fe | 7631, 7646 keV | 1.1 keV |
| $^{53}$Cr | $^{53}$Cr (n, $\gamma$) $^{54}$Cr | 835, 8885 keV | 4.2 keV |
| $^{58}$Ni | $^{58}$Ni (n, $\gamma$) $^{59}$Ni | 465, 8999 keV | 6.9 keV |

**Table 1:** Energies of prominent prompt $\gamma$ rays and 1st neutron resonances for light and medium elements.

The $\gamma$ ray spectrometer used for NRCA requires besides a high-energy resolution also a fast-timing response. The latter is needed since it is in an extreme $\gamma$ ray background originating from the presence of $^{137}$Cs in the debris. A study of the radioactivity in melted fuel of the TMI-2 accident by Uetsuka et al.

[8], reveals that the strongest activity originates from $^{137}$Cs, with a specific activity of ranging from $10^6$ to $3\times10^8$ Bq/g.

Furthermore, it is required to have a very high peak-to-Compton ratio, since the Compton edge energy for 661-keV $\gamma$ rays is very close to the energy of $\gamma$ rays induced by the $^{10}$B(n, $\alpha\gamma$)$^7$Li reaction. To satisfy these requirements, a well-type spectrometer made of $LaBr_3$ detectors was designed and is currently under development. A study based on Monte Carlo simulations of a well-type $LaBr_3$ spectrometer showed that the contribution of the Compton edge could be reduced by a factor by adding a back catcher detector. Such a reduction enables the identification of $^{10}$B even in the presence of a high background due to the decay of $^{137}$Cs [9].

## 3. Experimental Study

As a first experimental feasibility study of NRD, a normal shaped cylindrical $LaBr_3$ detector, whose crystal is 7.6 cm in diameter and 7.6 cm in length, has been utilized in NRCA experiments at the Geel electron linear accelerator (GELINA) facility of the EC-JRC-IRMM. The measurements have been performed with a $^{10}$B sample in the beam, with and without a standard $^{137}$Cs source attached to the detector.

**Figure 1** shows a pulse-height spectrum measured with the $LaBr_3$ detector for 478-keV $\gamma$ rays produced by the reaction $^{10}$B(n, $\alpha\gamma$)$^7$Li (shown in black colour). Figure 1 also shows a pulse-height spectrum (shown in red colour) for 661-keV $\gamma$ rays measured by the same $LaBr_3$ detector. The latter was deduced by subtracting from the spectrum recorded with both the $^{10}$B sample and the $^{137}$Cs source the one obtained with only the $^{10}$B sample present. It is shown that the 661-keV $\gamma$ ray peak due to the $^{137}$Cs decay is clearly separated from the peak corresponding to 478-keV $\gamma$ rays resulting from the reaction $^{10}$B(n,$\alpha\gamma$)$^7$Li. However, the Compton edge of the 661-keV $\gamma$ ray is very close to the 478-keV $\gamma$ ray peak. Therefore, the combination with the back catcher detector as discussed in chapter 2 is expected to have a significant role to identify the boron content in debris containing high radioactive $^{137}$Cs.

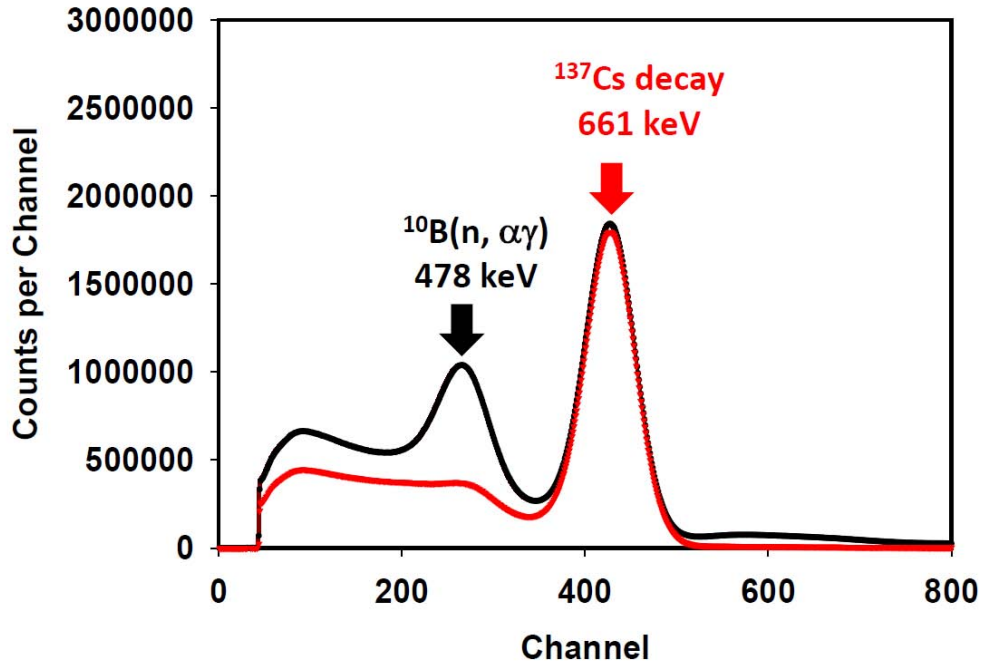


**Figure 1**: $\gamma$ ray pulse-height spectrum measured by a $LaBr_3$ detector

## 4. Design of Practical Facility

Based on a simulation study, the parameters required to design a NRD facility have been investigated. The minimum neutron flight path for NRTA to resolve resonances below 50 eV is 5 m. This parameter mainly determines the size of the facility. The size of a sample for NRTA is 10-30 cm in diameter and 1-2 cm in thickness.

A shorter neutron flight path is applied for NRCA, since elements are mainly identified by the energies of the prompt γ rays. Three beam lines are planned to be installed for NRCA, since the sample size for NRCA is smaller than that for NRTA. The size of a sample for NRCA is 1-2 cm in diameter and 1-2 cm in thickness.

The required intensity of the neutron source is in the order of $10^{12}$ 1/s. Such an intensity can be produced by a 1 kW electron beam with a kinetic energy larger than 30 MeV [11]. High energy neutrons produced by photonuclear reactions are moderated to epithermal neutrons by a moderator surrounding a neutron generation target. Neutron collimators are installed along the beam line for NRTA and the beam lines for NRCA. **Figure 2** shows a schematic view of the practical NRD facility including an accelerator room.

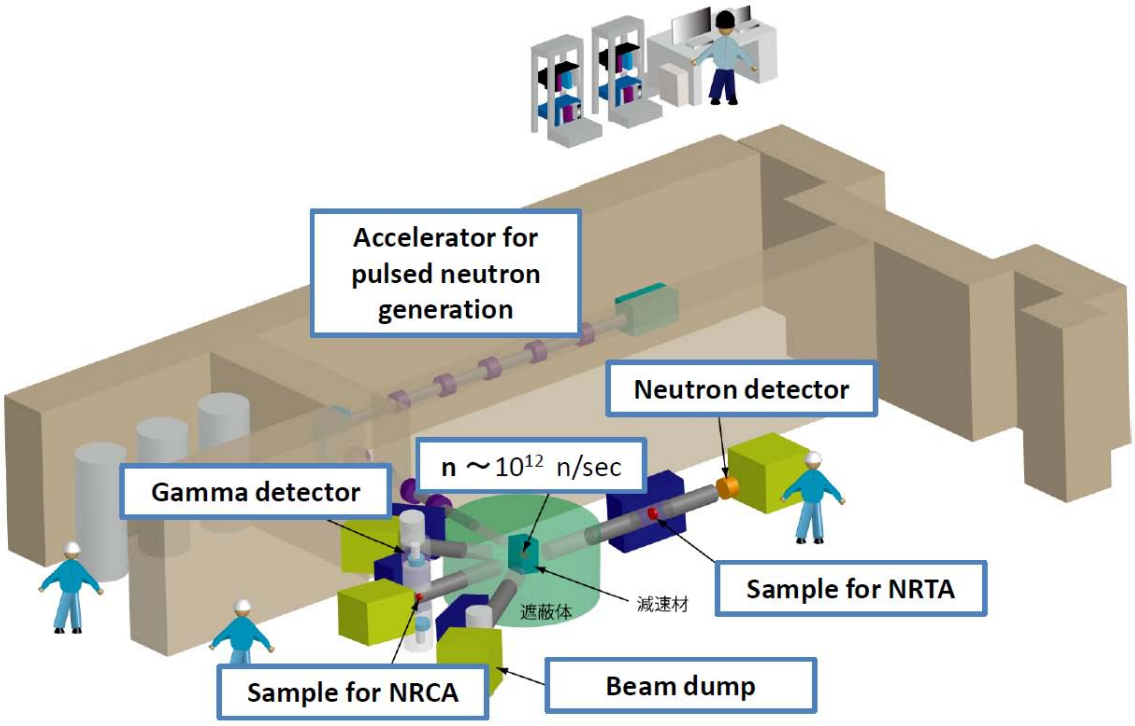


**Figure 2**: A Schematic View of the Practical NRD Facility

## 5. Evaluation of Uncertainties

The achievable uncertainty due to only counting statistics in case of NRTA was examined in Ref. [10]. The sample used in this study contained not only nuclear fuel materials but also matrix elements, such as B and Fe. The composition of the spent nuclear fuel was based on a burn-up of 40 GWd/t. **Figure 3** shows the achievable statistical uncertainty for each Pu- and U-isotope for a measurement period of 24 hours, a beam repetition rate of 100 Hz, and a neutron source intensity of $10^6$ 1/pulse (shown by red colour) or $10^9$ 1/pulse (shown by blue colour). In this case study 9 w% $^{56}$Fe and 2.5 w% $^{nat}$B were included in the sample. Th sample thickness was 1 cm and the diameter 30 cm.

By extrapolating the results shown in Figure 3, an uncertainty below 1% on the atomic number density due to only counting statistics can be achieved for the Pu and U isotopes with a measurement period of 20 min and a neutron source intensity above $10^{12}$ 1/s. This even under the condition that 9 w% $^{56}$Fe and 2.5 w% $^{nat}$B is present the sample.

A systematic study varying the sample thickness and the ratio of contamination materials has been carried out in ref. [10]. It was shown that the optimal sample thickness for NRTA depends strongly on the amount of contamination materials.

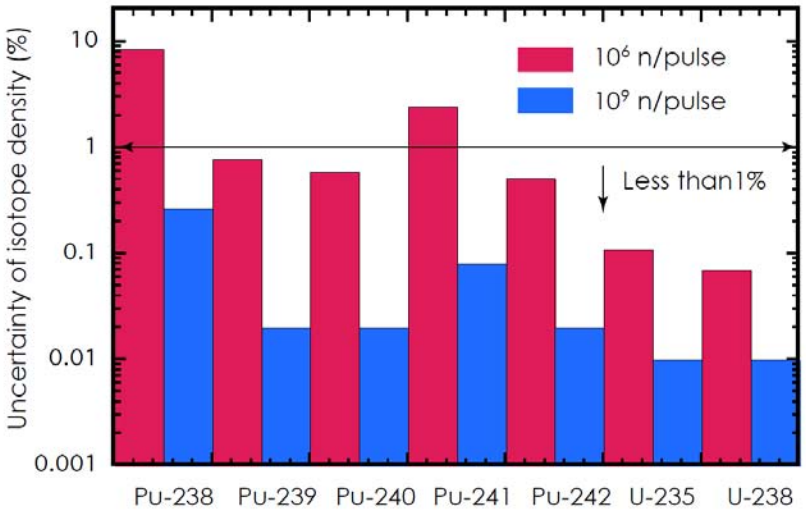


**Figure 3**: Achievable uncertainty due to counting statistics for U- and Pu isotopes for a measurement period of 24 hour, a beam repetition rate of 100 Hz, and neutron source intensity of $10^6$ 1/pulse (shown by red colour) or $10^9$ 1/pulse (shown by blue colour)

In order to evaluate the achievable accuracy and uncertainty of NRD, the next effects were identified to have a strong influence on the final results;

i) Particle size
ii) Sample thickness
iii) Presence of contaminated materials
iv) Sample temperature
v) The response of the TOF-spectrometer

To study the impact of these effects quantitatively, the resonance shape analysis code REFIT will be used. The code will be adapted to the needs of the NRD. Evidently, to quantify the amount of the Pu and U isotopes present in particle-like debris, the final accuracy will strongly depend on the quality of the resonance parameters used in the analysis, in particular, the parameters for the relevant Pu and U isotopes. Hence, a survey of the total cross sections is an important issue together with a measurement of some reference spectra.

## 5. Summary

We have proposed neutron resonance densitometry (NRD) to quantify nuclear materials in particle-like debris of melted fuel formed in a severe accident of nuclear reactors such as the Fukushima Daiichi nuclear power plants. NRD is a method that combines NRTA (neutron resonance transmission analysis) and NRCA (neutron resonance capture analysis) using a pulsed neutron generator and a neutron TOF (time of flight) technique.

One of the advantages of NRD to other non-destructive methods is its applicability to quantifying nuclear materials in a sample containing unknown components including strong neutron absorbing materials such as boron. NRCA is used to identify these elements included in melted fuel components. The other advantage of NRD is because NRTA, which is an absolute method, is applied to quantify the main elements (i.e. U and Pu). However, it requires a special treatment in case of inhomogeneous samples. A study based on simulations for melted samples containing $^{nat}$B and $^{56}$Fe showed that the

main Pu and U isotopes can be quantified in 20 minutes with a counting statistics uncertainty less than 1 % by using a neutron source of $10^{12}$ 1/s. This neutron intensity could be generated by a 1 kW electron beam having a kinetic energy larger than 30 MeV, which is produced using a small size electron linear accelerator.

A well-type gamma-ray spectrometer made of $LaBr_3$ detector has been designed for applying the NRCA under the circumstance of highly radioactive material mixture. Monte Carlo simulations showed that the identification of $^{10}B$ is possible within a 1 hour measurement even under the condition of strong background by $^{137}Cs$ using a pulsed neutron source with a source strength of $10^{12}$ 1/s. As initial experimental efforts, a normal cylindrical $LaBr_3$ detector has been used for NRCA experiments using a $^{10}B$ sample at the GELINA facility of EC-JRC-IRMM. It was demonstrated that the 478-keV $\gamma$ rays induced by the reaction $^{10}B(n, \alpha\gamma)^{7}Li$ can be clearly observed even in the presence of a strong $\gamma$ ray background due to the decay of $^{137}Cs$.

Further systematic studies are planned or under investigation for evaluating achievable accuracy accounting for uncertainties due to systematic effects such as the particle size, sample thickness, contaminated materials, and sample temperature and to demonstrate the effectiveness of NRD experimentally at the GELINA facility of EC-JRC-IRMM. Some recent advancements are also reported in this ESARDA meeting [9, 12, 13].

## 6. Acknowledgements

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# Images Objects vs. Pixels: A comparison of new methods from both domains

**Sven Nussbaum[1], Julia Tueshaus[1], Irmgard Niemeyer[2]**

[1]University of Bonn, Geographic Institute, Center for Remote Sensing of Land Surfaces (ZFL), 53113 Bonn, Germany,
sven.nussbaum@uni-bonn.de, julia.tueshaus@uni-bonn.de
[2]Forschungszentrum Juelich GmbH, Institute of Energy and Climate Research, IEK-6: Nuclear Waste Management and Nuclear Safety, 52425 Juelich, Germany,
i.niemeyer@fz-juelich.de

***Abstract:***

*Under the Additional Protocol of the Non-Proliferation Treaty (NPT) complementing the safeguards agreements between States and the International Atomic Energy Agency (IAEA), commercial satellite imagery is an important source of information within the "information driven safeguards" approach of the IAEA. The new Geospatial Exploitation System (GES) creates a huge demand for spatial information and new tools to analyze this data. The IAEA faces the challenge of a steadily increasing number of nuclear facilities worldwide and therefore (semi-) automated and computer driven methodologies can add a big value in the verification process. Another challenge of the IAEA is to stay on-top of new technologies and to use them effectively and efficiently within the safeguards verification process. In the last decade several new pixel-based and object-based approaches for image analysis and for change detection have evolved and this paper aims at evaluating and testing some of them in the field of safeguards. New pixel-based classification algorithms like the support vector machines or time series analysis have been widely used in the field of land cover change analysis, but haven't been applied in the verification process. New developed object-based change detection approaches and optimized workflows have been developed and should now be tested in the field of safeguards. The aim of this paper is to apply and evaluate some of these methods using the example of different nuclear sites, and to compare their suitability in the context of information-driven-safeguards.*

**Keywords:** satellite imagery, change detection, time series analysis, object based image analysis

## 1. Introduction

*"Spatial is special"* or "*Think spatial!"* these mottos seem to have arrived in the International Atomic Energy Agency (IAEA). In concert with the paradigm change from traditional to information-driven safeguards, geoinformation has a significantly increased importance in the safeguards analysis process. One key reason for this development is the implementation of the new Geospatial Exploitation System (GES). With the GES geoinformation (GI) like satellite imagery or site plans of nuclear facilities are available for a broad range of inspectors, analysts or country officers. As soon as this wider audience will gain access to that kind of information, they will recognize how valuable it is, and the request for geoinformation will continue to grow.

The increased demand for spatial information goes hand in hand with a steadily rising number of nuclear facilities under (integrated) safeguards worldwide. Here, computer-driven processing algorithms and tools become essential for automating the analysis of the vast amount of safeguards-related data as much as is reasonably possible. In particular, algorithms for pre-processing satellite imagery acquired by newly sensors or geoprocessing models that generate additional safeguards relevant information, could add a big value in the safeguards verification process, for examples see Nussbaum & Niemeyer [1], Niemeyer et al. [2] or Listner & Niemeyer [3]

With the improving spatial resolution and the amount of details visible on remote sensing data, also new algorithms for analysing very high resolution satellite imagery with regard to safeguards verification are needed. Object-based approaches show promises against pixel-based one as they try to imitate the image understanding of an image analyst, see e.g., Listner & Niemeyer [4,5], or Nussbaum & Menz [6]. An image analyst can easily identify image regions along with their colour, shape, texture and context, and categorize them into objects of interest, such as buildings, streets, forests etc. Computer-driven object-based image analysis is an approximation to human perception. Using specific object features (e.g. colour, shape, texture or context) of defined object classes through rule bases, each object can be assigned to an object class. The selection of the optimum features can either be performed automatically by applying statistics to a set of samples for each class or interactively by integrating knowledge on the object classes in the recognition process. Object-based image analysis approaches are far from being able to perform an image interpretation on its own, however, they offer some time and cost effective procedures for object recognition and feature extraction, especially when using very high resolution satellite imagery.

This development towards object-based methodologies follows a trend recognized by Blaschke [7, p.12] already in 2010. He concluded from a comprehensive literature review that OBIA represents a significant trend in remote sensing and GI science. With this new trend also the availability of new tools in the area of OBIA has increased. This paper presents a workflow which enlarges the possibilities of OBIA with adapted tools in the area of segmentation und feature analysis. By including these tools the analysis process itself can be made more straightforward and time saving.

However, also advanced pixel-based approaches for image classification, such as support vector machines algorithms, see e.g. Mountrakis et al. [8], are still worth considering. As far as low or medium resolution satellite data is concerned, time series analysis (TSA), see e.g. Lambin & Linderman [9] could be another possible source for (mostly free) valuable safeguards relevant information. TSA analysis is applied to a huge sequence of satellite images acquired over the same area of interest in order to detect trends and changes over time.

This paper aims at discussing some workflows, novel algorithms and methods to be applied in the field of safeguards verification. A comprehensive evaluation or detailed case studies have not been performed yet, thus the quality of the tools will be assessed visually. Moreover, their user-friendliness and easiness and the time exposure will be discussed. In summary, the authors are confident that these novel tools can produce safeguards relevant information and support the verification process.

## 2. Image Classification

The aim of image classification is to determine areas or objects of interest in an (semi)-automated way. Objects of interest can be, e.g., buildings, pipelines or other safeguards relevant structures in a satellite scene. While discussing both the object-based and the pixel-based approach, the paper focuses on the improvement of segmentation and feature analysis in the OBIA workflow.

Digital image processing in remote sensing usually follows a certain workflow. The first general step is usually the pre-processing of the imagery. This step is necessary for both pixel- and object-based image analysis and includes geometric correction and/or atmospheric/radiometric correction. Since pre-processing does not differ for either domain, it will not be further discussed here.

In OBIA, the step following the pre-processing is the generation of image objects. This can be done using a variety of existing segmentation methods. The given study uses the so-called multi-resolution segmentation implemented in the software eCognition 8.7, which has proven to achieve reasonable results, see Marpu et al. [10].

Segmentation has always been one of the most crucial parts, as it bears the problem of correct parameterization in order the retrieve meaningful objects that are close to real-world objects. So as to cover all objects with their varying sizes and relations, a hierarchical network of objects with different object levels representing smaller, mid-scale and bigger real world objects is needed.

One parameter is the so called *scale parameter* and it determines (simply said) the size of the resulting image objects. In most studies scale parameters have been found using a time consuming

and unreliable trial-and error approach. Dragut et al. [11] proposed a statistical method in order to determine the best scale parameters for the multiresolution image segmentation. The estimation of scale parameter (ESP) tool builds on the idea of local variance of object heterogeneity within a scene. The ESP runs as a process in eCognition and generates multiple hierarchically linked images object levels. The local variance is calculated for each scale level and plotted against the corresponding scale. The peaks in the graph at which the local variance changes indicate the most appropriate scale levels based on the data properties of the scene [11].

Once the image objects have been generated, the next step in OBIA is the definition of the rule sets for classification. The advantage in OBIA of having much more possible feature at hand (colour, shape, texture or context) to classify can turn into a disadvantage in the sense of finding the correct features to use for classification. How to choose the best features from the list of over hundreds possible feature descriptions in eCognition? Which features are the typical ones for this or that class, which are the ones the separates the classes' best? Nussbaum et al. [12] proposed a statistical tool SEaTH in order to calculate the *Separability* for each class combination for any number of features. The tool also determines the *Threshold* where classes can be separated best. The implementation of this tool in eCognition can be downloaded at: www.treatymonitoring.de/tools

From a users' perspective these tools are easy to handle and they make the OBIA workflow fast and more straightforward (see Figure 1).

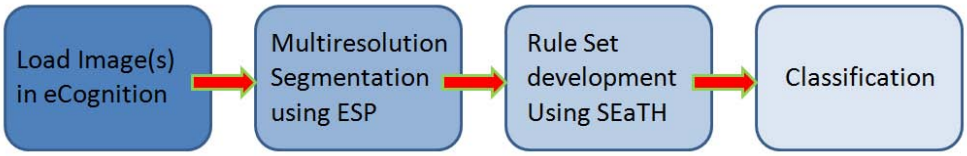


Figure 1: OBIA workflow using additional tools

Figure 2 a) shows as test area a nuclear facility. A subset of a QuickBird image from August 2005 to be used as test image shows a small part of the facility with buildings, streets and background (soil) which represent the target classes of interest. According to the OBIA workflow the image was loaded into eCognition and then segmented by using the ESP tool [11]. Figure 2 b) shows the interface of the ESP tool; here, the grey circles indicate peaks of local variance, i. e. possible good scale parameter settings for segmentation. The multi-resolution segmentation generated three hierarchical object levels using the local maxima of the scale parameter at 80, 150 and 220.

After segmentation the samples for the classes *buildings, streets* and *soil* where defined in eCognition. These samples where used in the SEaTH tool [12] to determine the best features and thresholds for separating the classes of interest. Figure 2 c) shows the interface of SEaTH, which highlights as a result the best features and thresholds to choose. Based on the SEaTH output the class descriptions where done, followed by the classification process itself. The result of the classification is shown in Figure 2 d).

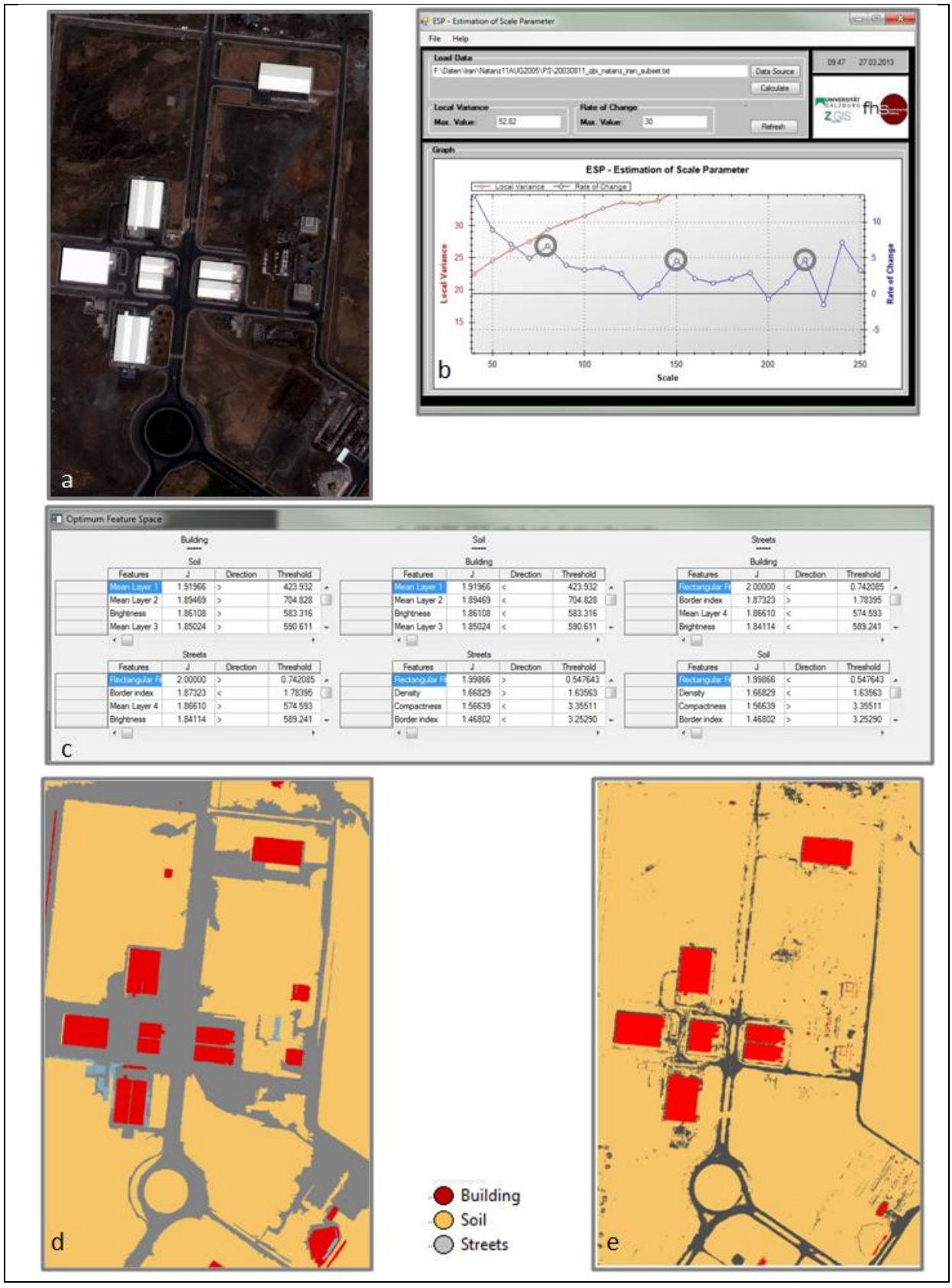


Figure 2: OBIA workflow and comparison of object-based and pixel based classification. a) Test image: subset of QuickBird image over a nuclear facility, copyright DigitalGlobe; b) ESP interface and results; c) SEaTH interface and class features; d) result of OBIA classification; e) result of support-vector machine pixel-based classification.

A brief visual validation showed that all the big buildings were detected and that the streets seemed to be a little over-classified.

In comparison, the support vector machine (SVM) classifier implemented in the ENVI software package was applied for pixel-based classification. After loading the image and the definition of class samples to calibrate the algorithm, the classificatory runs fully automatically. The result of this is shown in Figure 2 e). The known salt-and-pepper effect for pixel-based algorithms using high-resolution imagery is still visible in the results. However, the main streets show more realistic shape compared to the OBIA street class, and the majority of the big buildings have been identified. The SVM misses 4 buildings compared to the OBIA. This is probable due to the fact that these four buildings are darker compared to the others and are thus not considered for the SVM building class which is solely based on the spectral features.

## 3. Change detection - Time series analysis using medium resolution imagery

One major focus in the development of change detection (CD) methods for safeguards applications was the implementation of object-based techniques focussing on the change between two acquisition times [3]. Since 1972, satellite sensors with low to medium spatial resolution were launched in order to continuously acquire surface data. Therefore the analysis of large image series is possible. Since the Landsat archive was opened to public in 2008, images with moderate-temporal and moderate spatial resolutions (up to 15m) have become freely available. Accordingly, time-series analysis (TSA) of remote sensing data has advanced. Instead of comparing two images from two different dates, TSA analyses plenty images (up to hundreds) acquired over the same area. The idea is to study the change of every pixel in the time series through time and see how each pixel evolves. With this development, trends over time, seasonal or abrupt changes can be made visible, see Verbesselt et al. [13]. Because many images are needed, most studies use freely available medium resolution imagery like Landsat, for examples see Dubovyk et al. [14, 15].

But how could TSA using medium resolution satellite imagery be applied for safeguards verification? One scenario in which TSA could play a role is the detection of construction activities at former clandestine sites. As long as sites were clandestine, usually no commercial operator of VHR sensors would task this particular region for image acquisition, thus the operator might not have any data from the start of construction in the archive. Maybe, the construction started before the first VHR satellite was launched at all. Given the absence of data for the start of construction, this particular point in time could be narrowed down by using a semi-automatic TSA trend analysis. We used a former clandestine underground site for investigation.

Figure 3: Google earth imagery ©,

Figure 3 shows a Google Earth time-series. VHR imagery before 2004 were neither available from Google Earth nor from the Digital Globe Archive. Our study focused on the two buildings and the construction activities like streets between 2000 and 2004. This site was chosen despite the fact the national government claimed that at that time there were no nuclear activities at the particular site.

For the given site all available and cloud-free Landsat-7 imagery between 2000 and 2004 were used. This paper only shows the results for 2002, as this is the year when the construction activities were detected. For the year 2002 16 images could be downloaded from http://earthexplorer.usgs.gov/. For

pre-processing the data, a relative radiometric correction was applied to the images using the IR-MAD algorithm; see Canty & Nielsen [16].

We used TSA to automatically generate a trend profile for our Area of Interest (AoI) for the Landsat series. If a strong trend existed, the area of the trend was masked out, and a temporal profile for that area was generated. In this temporal profile the peaks or troughs of the curve determined the dates of maximum change and could be directly linked to the date of the images. Therefore an image analyst could simply focus on these dates and visually check the change (e.g. construction start) at the AoI.

Using the Earth Trends Modeler software[1] a trend from these 16 images was calculated. For trend calculation, the pixel value for every pixel per band was plotted against time and the trend was calculated by a linear regression per pixel over time. In our case we choosed band 3 as our major indicator, because the spectral difference between built-up area and surrounding background maximizes in this band, see Figure 4. In principal, any other band, band combination or index could be used.

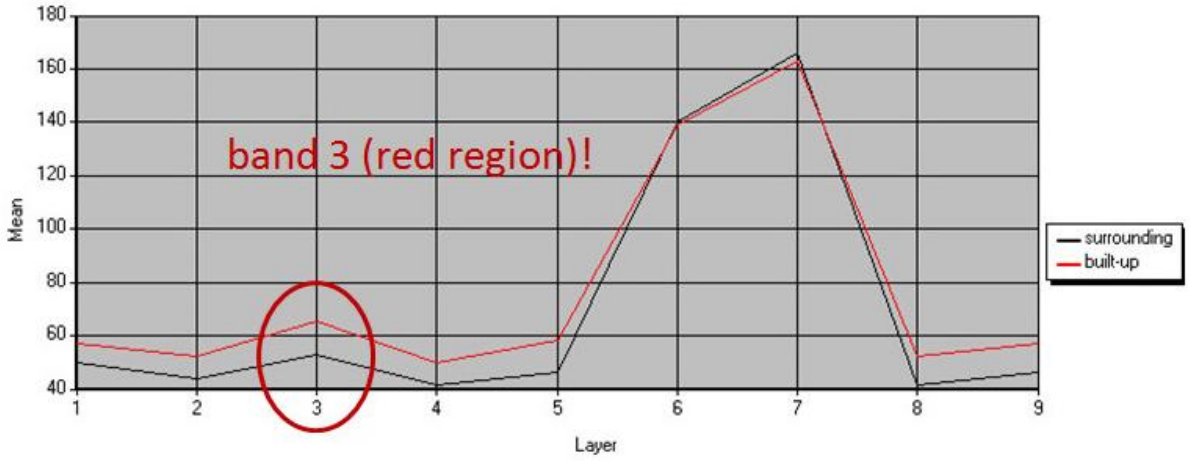


Figure 4: Third band is a major indicator for change in the trend analysis

The trend image shown in Figure 5 indicates a huge change within our AOI (strong positive trend) in 2002. In order to detect only significant changes a threshold was applied, in our case three standard deviations from the mean trend value. The areas which are within this threshold were then masked out (Figure 6a) and a temporal profile for that mask was generated (Figure 6 c). By superimposing the masked area on a Landsat scene (Figure 6 b) for visualisation purposes, it turned out, that the mask automatically generated by the threshold includes the area of the two buildings.

The temporal profile clearly shows that there was a big change signal between 25 March 2002 and 05 May 2002. The image analyst can now focus on these time period and visually check the images. Both images are shown in Figure 7. It is obvious that the construction activities start in between these two dates. Thus, in this case study we were able to narrow down the construction start date to a time-frame of 40 days accuracy.

[1] Copyright by IDRISI ©

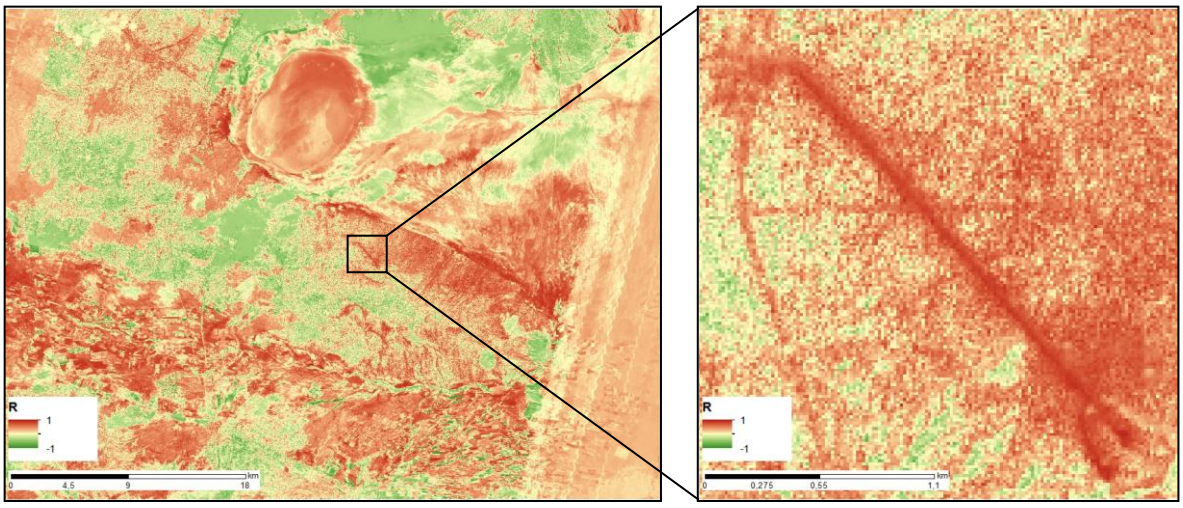


Figure 5: Trend image of the AoI at the former clandestine site for the year 2002

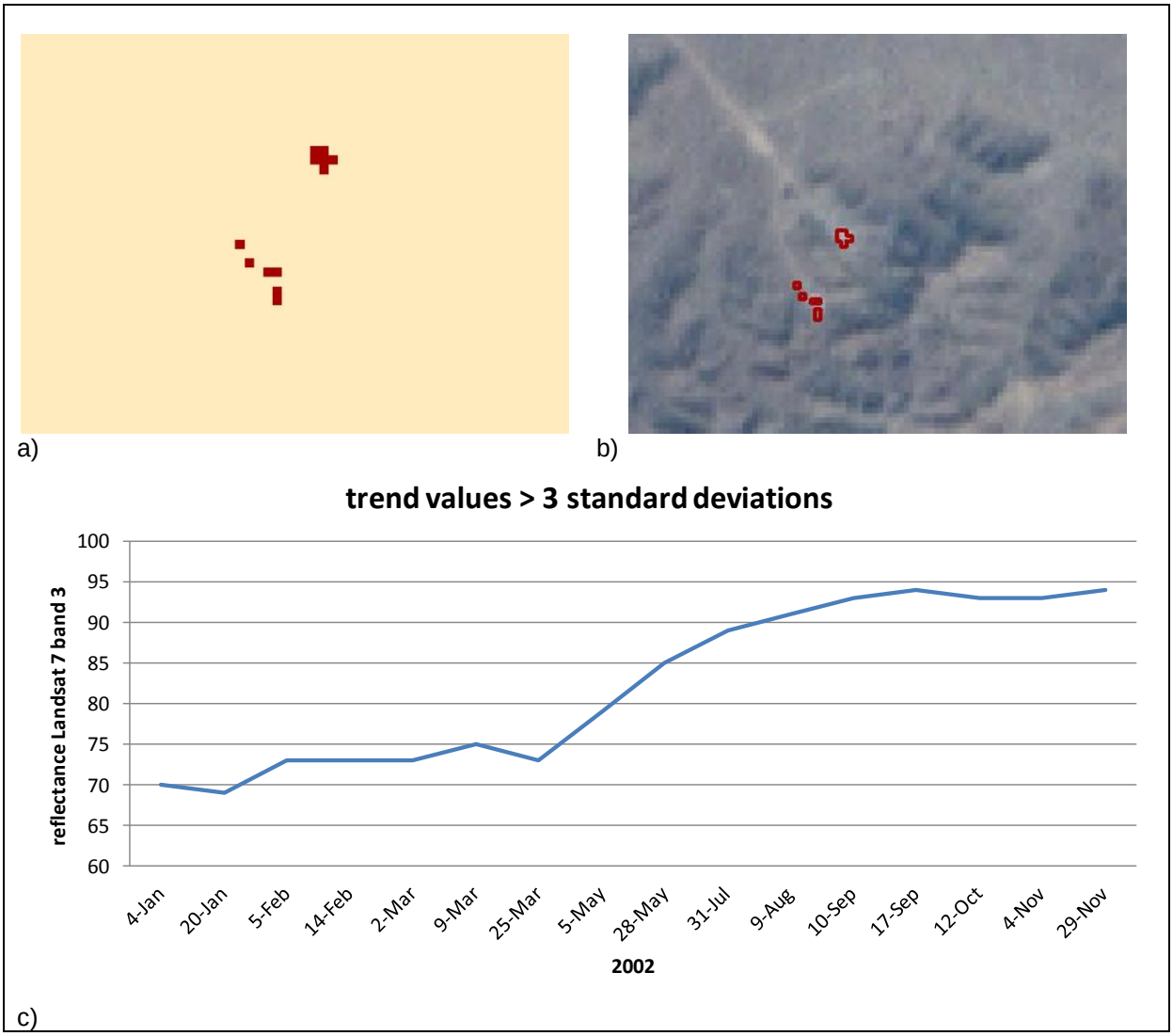


Figure 6: Masked out area of significant change (a,b) and temporal profile for the masks (c).

Figure 7: Visual comparison of the detected construction start by the methodology.

## 5. Conclusions

In order to cope with the challenges of information driven safeguards and geoinformation the IAEA needs to stay on top of new technologies. Thus it is very valuable to consider new analysing techniques or tools. It is also important to investigate in existing techniques which have not been applied to safeguards yet. In chapter 3 new tools for streamlining the OBIA workflow have been applied and implemented. It could be shown that these simplify the OBIA workflow and close some of the gaps which still existed. These gaps where segmentation and feature analysis. When using the tools ESP and SEaTH in combination it makes the analysis process much easier and straightforward. A comprehensive validation is out of the scope of this paper. The aim was to show that by using additional tools in the OBIA workflow, it becomes easier, faster and also straightforward. The limitation is that the tools used were not integrated into one software package and that results had to be manually transferred between different programmes. This makes it not suitable for the IAEA analyst right now, but in principal OBIA is not far from being operationally useable. As soon as all add-ons and tools would be implemented into one software package, e.g., eCognition, they could be used in the analysis process at the IAEA.

Testing and applying new pixel-based and more robust classifiers like the support-vector machine algorithm leads to comparable results, but here also a more extensive validation should be carried out. Shortcomings in terms of misclassification also demonstrate the advantage of the OBIA approach, which in our example considered a shape and a spectral feature to identify the buildings. Nevertheless new pixel-based classifier like the SVM should still be considered when analyzing high-resolution imagery since they enhance and become more powerful too.

Concerning change detection, time series analysis has shown to be a valuable tool in the field of safeguards. With our methodology it is possible to (semi-) automatically narrow down construction activities of sites based on medium resolution imagery. This can be very helpful in the case when there are no high resolution images available because the activities where clandestine. It can also be applied to sites where construction or other activities happened in the past, during a time when no commercial high-resolution sensors where available at all. For the future, this methodology will be further investigated and applied to different case studies. It will be tested which bands, band combinations or indices can improve the change signal. Nevertheless, for our case study, the results were reliable and generated safeguards relevant information.

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# Ahaus remote data transmission (RDT) field test – from the operators' point of view

**A. Jussofie[1], W. Graf[1], K. van Bevern[2], I. Niemeyer[3], A. Rezniczek[4], W. Trautwein[5], K. Schoop[6]**

[1]Gesellschaft für Nuklear-Service mbH, Frohnhauser Str. 67, 45127 Essen, Germany
[2]VGB PowerTech e. V., Klinkestraße 27 - 31, 45136 Essen, Germany
[3]Forschungszentrum Jülich GmbH, IEK-6: Nuclear Waste Management and Reactor Safety, 52425 Jülich, Germany
[4]UBA Unternehmensberatung GmbH, An Gut Forensberg 40, 52134 Herzogenrath, Germany
[5]Bundesministerium für Wirtschaft und Technologie Villemombler Str. 76, 53107 Bonn, Germany
[6]European Commission, DG ENER E1, L-2920 Luxembourg

E-mail: astrid.jussofie@gns.de, katrin.vanbevern@vgb.org, i.niemeyer@fz-juelich.de, Rezniczek@uba-gmbh.de, wolfgang.trautwein@bmwi.bund.de, Konrad.Schoop@ec.europa.eu

***Abstract:***

*The prolongation of inspection intervals due to the implementation of the Integrated Safeguards (IS) regime involves the need to improve the effectiveness and efficiency of the inspection efforts in nuclear facilities. The application of remote data transmission (RDT) for review and evaluation of equipment state of health (SoH) data and safeguards data (including images) is becoming increasingly important for EURATOM and IAEA. However, one of the main operators' concerns about remote transmission of safeguards data being collected by unattended containment and surveillance (C/S) systems to EURATOM headquarters is the risk of an unauthorized data access by third parties. Furthermore, it has to be ensured that the original SoH and safeguards data reach their destination without tampering and that the sender's identity can be authenticated.*

*A field demonstration test at the central dry storage facility at Ahaus was initiated as a task of the German support programme to the IAEA to provide evidence that the RDT system will meet the expectation of all stakeholders involved. In order to build up confidence and experience, the field test was carried out in two steps firstly the transmission of equipment state of health data and seal statuses followed by the transmission of sensitive image data in the second step.*

*The paper summarizes the essential preparations, which started with an on-site inspection in January 2011 followed by a technical meeting in order to determine how to handle the sensitive image data. The paper discusses the determination of boundary conditions for transmitting Safeguards data remotely to EURATOM and presents the findings and experiences gained from the field trial from the operator's point of view.*

**Keywords:** Remote data transmission; field trial; facility operators' experience

## 1. Introduction

With the implementation of Integrated Safeguards in Germany on 1st January 2010 the quarterly routine inspections in light water reactors (LWRs) and spent fuel storage facilities (SFSFs) were replaced by short notice random inspections (SNRIs) with a selection probability of 20% per year. Based on the existence of 16 dry SFSF (including 12 on-site dry storage facilities, the two central storage facilities Ahaus and Gorleben, the storage facility at Lubmin and the storage facility at Jülich) with more than one SQ not more than 4 SNRI per year could be expected. That means that SFSF are often only inspected once per year for Physical Inventory Verification. In view of these long and varying inspection intervals it is essential to ensure the functionality of the Safeguards equipment in use. Since emphasis is put on the short inspection notification time of 24 h (recently extended for the first time to 48 h), automatic and unattended monitoring during the notification period is of particular

importance in the Integrated Safeguards (IS) regime. Currently, safeguards data are stored compressed but unencrypted on removable hard disks which the inspectors transport in their briefcases to Luxembourg or Vienna. Therefore one objective was to make data transmission more secure than before. Besides, EURATOM favors a timely transmission and verification of equipment state of health (SoH) data in order to be able to respond fast to a functional loss or malfunction of safeguards instrumentation such as surveillance cameras and thus to minimize the risk of a re-verification. The IS related modified inspection regime has led a reduced presence of IAEA inspectors. In Germany without a national safeguards authority, EURATOM is in charge of establishing and maintaining the state's system of accounting for and control of nuclear material subject to safeguards under the agreement INFCIR/193. In compliance with this obligation, EURATOM can only reduce their inspection frequency with the application of compensating measures in order to keep the quality of their control function. Therefore, EURATOM as well as the IAEA consider the application of remote data transmission (RDT) for daily monitoring of equipment state of health (SoH) data and safe transport of safeguards data (including images) to be suitable to reduce their inspection effort on-site thereby achieving the desired saving of personnel and financial expenses.

The German operators have always understood safeguards as confidence-building measures to the international as well as national public. By a smoothly running safeguards system the operators can show the public that even at the international level the German nuclear power plants are evaluated as proliferation-safe. The fact that the measures applied by the operators are consistent with the requirements of EURATOM and IAEA and performed without complaints by them is also of great importance for the federal Government because ultimately the state has to prove credibility with regard to the IAEA commitments.

Based on this belief, the German operators have always supported the safeguards measures within their means and thereby contributed to an improvement of the transparence of safeguards measures. Accordingly, the operators have also been involved in the latest modification of the safeguards regime associated with the implementation of IS. In case of RDT the operators have seen the benefits for EURATOM and IAEA on the one hand but expressed their reservation on the remote transmission of images on the other hand. The operators' main concern about remote transmission of safeguards data being collected by unattended containment and surveillance (C/S) systems to EURATOM headquarters is the risk of an unauthorized data access by third parties. Furthermore, it has to be ensured that the original SoH and safeguards data reach their destination without tampering and that the sender's identity can be authenticated.

The implementation of RDT will not be effective if the requirements to protect the sensitive information are specified on a facility by facility basis. Based on the federal structure in Germany most of the facilities have different license and supervising authorities. According to the operators, the responsibility of ensuring the existence of a sufficient security of sensitive information lies with the federal government of Germany. It is necessary to determine the essential boundary conditions for transmitting sensitive safeguards data remotely from the facility to the EURATOM headquarter in order to ensure the same level of information security across all the facilities of a specific type of installation.

The proposal of the Federal Ministry of Economy (BMWi) to perform a field trial of RDT under the German Support Programme reached a broad agreement among the parties involved BMWi, BSI (Federal Agency for Security in Information Technology), FZJ (Forschungszentrum Jülich), WKK (Nuclear Fuel Cycle Association) and VGB (Association for Power and Heat Generators). For this purpose the central spent fuel dry storage facility at Ahaus was selected. Here, the infrastructure of a previous field test was already available and the control of nuclear material was supposed to be easier than in the decentralized interim dry storage facilities close to the reactor, since no changes of the nuclear inventory are currently taking place at Ahaus.

The aim of the field test was to select the technical components for the remote transmission of safeguards data taking into account the German security requirements for sensitive data and to demonstrate the practicality of the RDT concept developed by EURATOM and modified according to the recommendations of the BSI under daily use conditions. The focus of the field trial was to generate and establish a full-functional solution. For this, it was necessary to work out an administrative and technical solution for the RDT from the transport cask storage (TBL) at Ahaus to the EURATOM headquarter at Luxembourg under the general guidance of the Federal Ministry of Economy. The administrative solution has been determined in the boundary conditions for the field trial of RDT. The recommendations of the BSI have been implemented in the principal technical solution based on the concept of EURATOM and IAEA.

## 2. Requirements to the RDT system from the operators' point of view

Since the German interim dry storage facilities are equipped with essentially the same safeguards instrumentation as the interim dry storage facility at Ahaus, a further aim of the field test was to develop boundary conditions, which are transferable, preferably without significant changes, to all the German dry storage facilities. The most important technical requirements to the RDT system from the operators' point of view are as follows:

- RDT transmission via a separate communication line:
  For the operators it is crucial that RDT shall not have an impact on the plant operation. Besides, RDT is to exclude the transmission of operating data as well as the monitoring of personnel and plant operation. For this reason the operators regard a communication that is separated from the operator network as mandatory for RDT. From the operators' point of view the RDT without using the company's own data network is the easiest and most effective measure to protect the operator network from the introduction of malicious software when implementing RDT.
- Delayed image transmission:
  The acceptance of the image transmission with a delay of 24 h is important for the operators because it gives the operators the opportunity to be on the same level of information as EURATOM and IAEA. On the other hand a delay of 24 h represents no impairment in the routine image transmission for EURATOM and IAEA. In contrast, information on the functional status of the involved technical compounds and image-free safeguards data might be transferred without delay.
- Secure transmission of image data:
  A differentiation between state of health data for monitoring of surveillance equipment and functional states of the electronic seal on one hand and image data on the other hand is of fundamental significance for the operators because during the optical surveillance of a material balance area sensitive data are generated which may affect the physical protection of the plant or economic interests. The proof of the function of surveillance cameras, electronic seals and possibly neutron detectors by the transmission of SoH data (functionality of the device) for the period between inspection notification and inspection performance is a prerequisite for a SNRI under IS. Efficiency gains under IS substantially depend on the condition that no re-verification of the nuclear inventory will be required. Therefore, a transmission of SoH data is also in the interest of the operator. Compared to SoH data the transmission of sensitive image data calls for higher secure requirements for the RDT system in order to counteract the growing threat of targeted attacks to spy and manipulate data.
- No increase of the operators' effort by RDT:
  With RDT an effort reduction should be achieved on the part of the operators as well as on the part of EURATOM and IAEA. Therefore the inspection frequency should not be increased. This assumes that the RDT system works reliably under daily use conditions and the Safeguards data are transmitted completely.

## 3. The concept for RDT

Safeguards data may contain sensitive information which shall not be accessible to unauthorized parties. Consequently, safeguards data were divided into two categories:
The first category included State of health data (information on the functionality of the device) and Safeguards data without images such as the status of electronic seals. The second category consisted of images requiring additional security measures which had to be taken into account in the concept for RDT. The principle of RDT system developed by EURATOM and IAEA is the establishment of a VPN (virtual private network) tunnel between the data acquisition system in the dry storage facility at Ahaus and the data server in the EURATOM headquarter at Luxembourg using a broadband internet connection which is separated from the operator's IT network. At the beginning of the VPN tunnel in the TBL-Ahaus and at its end in the EURATOM headquarter there is a VPN router with a firewall and an encryption/decryption function. The remote VPN device establishes the secured tunnel by knowing the HQ IP address, whereas the pull of all data is triggered by an isolated and well secured server at Luxembourg. Following the transfer of data to Luxembourg the data are immediately transmitted to Vienna via a further VPN tunnel.

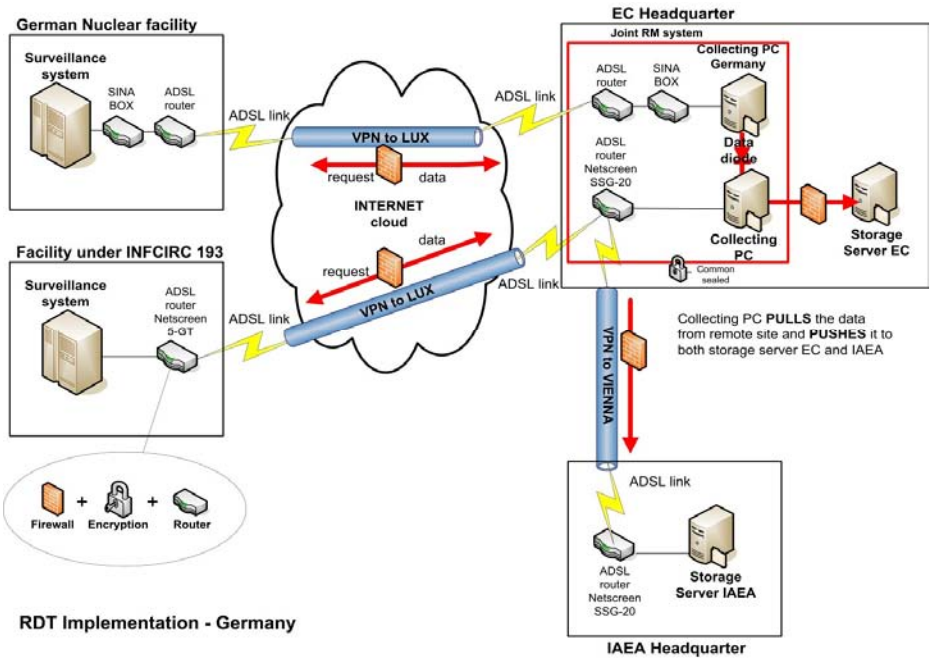


**Figure 1:** Concept for RDT implementation in Germany

Due to the concern of the operators regarding the security of sensitive safeguards data EURATOM offered a modification of the RDT system described above in order to take German security requirements for sensitive image data into account. The development of a basic technical solution for the RDT implementation in Germany was carried out in close coordination with the Federal Agency for Security in Information Technology (BSI). Two additional measures were recommended by the BSI and implemented in the original concept by EURATOM:

a. The application of the SINA-technology, which is approved by BSI for the security classification VS-nfD (Germany), EU restricted and NATO restricted, ensured a secure handling of image data.

b. The implementation of a data diode (SINA OWG2 Gateway with input and output server) in the RDT system guaranteed a unidirectional data transfer. For this purpose the data collecting PC was doubled, so that the data diode was added between the PC collecting the Safeguards data from Germany and the PC collecting the data from other non-weapon states of the EU. By decoupling the German RDT-network from other RDT networks in this way a potential access from installations outside Germany to EURATOM IT systems of installations in Germany could be excluded.

During data transmission via the VPN tunnel generated by the two SINA-boxes, one installed at Ahaus and the other one at Luxembourg, the Safeguards data are encrypted according to the Advanced Encryption Standard (AES)-algorithm by a 128 bit key which is only known by the two SINA-boxes. The PPPoE protocol is ensuring that during the transmission over the internet all data packages from the sender will arrive at the receiver without loss. All images are signed with an authentication signature, calculated already during the image capturing process in the DCM14 camera module. This signature is automatically verified during the review in the inspector's headquarter to ensure the genuine of the image.

## 4. RDT field trial

### 4.1. Test content

The field test comprised two phases. The first test phase was limited to the transmission of SoH data of the camera and the EOSS which was opened and closed for test purposes by the operator at Ahaus. The times were recorded for comparison with the signals received at Luxembourg. The EOSS

was installed close to the EURATOM/IAEA cabinet outside the storage hall. The second test phase extended to the transfer of sensitive image data.

### 4.2. Preparatory activities

The administrative solution is mainly based on the boundary conditions for the field trial of RDT which have been agreed between EURATOM, BMWi, BSI, FZJ , VGB and finally with IAEA.

The essential preparatory activities for the development of a technical solution adapted to the German security requirements started with an on-site inspection in January 2011 to check the available infrastructure and further needs, followed by, amongst others, a technical meeting with the BSI in February 2011 to discuss the handling of sensitive image data. A list with transmission devices certified by the BSI was provided in April 2011 to EURATOM. After consultation with the IAEA few SINA boxes and the data diode were purchased. After tests of the equipment and completion of its set up in Luxembourg the RDT-components were installed at Ahaus on 12.06.2012; the internet broadband connection was activated on 21.06.2012. The SINA box requires a detailed programming of all network structure with IP addresses which can only be done in Luxembourg. The test was performed for the configuration of a SINA box connected to an ADSL router but the operator of the TBL-Ahaus had ordered a SDSL line with a modem based on his existing maintenance contract. Therefore the first attempts to connect both SINA boxes were not successful. A stable VPN-tunnel created between the SINA boxes at Ahaus and Luxembourg was achieved for the first time on 6.09.2012.

### 4.3. Results

In the first phase of the field trial the focus was laid on verifying the availability and completeness of the remotely transmitted data by checking the number of the transmitted SDIS-log-files and comparing the number of seal openings and closings with the readout at Luxembourg. In the period from 24.09.2012 to 25.10.2012 the EOSS was opened and closed 33 times, respectively. The registration of 66 signals at Luxembourg demonstrated that the remote transmission of seal data was complete.

As a result from this positive outcome the second phase of the field trial started with the transmission of original images from the four surveillance cameras installed in the interim dry storage facility at Ahaus, taking the afore mentioned boundary conditions into account. A preliminary phase of transmitting test images was waived. Based on the daily transmitted image data file generated by the SDIS system EURATOM confirmed the completeness of the remotely transmitted images. It could be demonstrated that all the images were genuine, recorded and transmitted with the valid (authentic) signature and the correct recording interval.

Thus the results of the field trial showed that the three main attributes of sensitive information as confidentiality, integrity and availability can be protected by the security measures implemented in the RDT-system. EURATOM confirmed the complete transmission of the locally stored image data.

In view of the possibility of a data loss due to an interruption of the RDT operation, also the option for a retroactive transmission of image data was verified in the field trial. This was achieved after the start of the image transmission on 7.11.2012 by the subsequent tracing of the transmission of older images back to the date of 10.10.2012. The subsequent transmission of the accrued image data led to a utilization of the RDT-system for several hours. Therefore it was shown that the backup of the redundant locally stored data to the remote server at Luxembourg works.

## 5. Conclusion

In summary, the RDT field test at Ahaus which lasted over several months has been evaluated as successful by all parties involved. From the operators' perspectives the requirements determined in the technical boundary conditions can be met with the technical solution gained by the implementation of the recommendations of the BSI in the original EURATOM concept for RDT. EURATOM expectations have been fulfilled because the RDT system provided a complete and reliable data transmission applicable under daily use conditions. Through the application of security technologies certified by BSI the operators' main concern of a risk of an unauthorized access to safeguards data by hackers via the internet as well as the potential access of nuclear installations located outside Germany to the EURATOM IT-systems in German installations was taken into account. Besides, interruptions of the RDT-operation by several days remained without consequences for both sides due to the local storage capacity of the RDT-system and its subsequent automatic synchronization.

# Experiences in Uranium Abundance Analysis Using the MTE Methodology

***C. Black, E. Zuleger, J. Horta-Domenech, M. Vargas Zuniga***

Institute for Transuranium Elements
Joint Research Centre, European Commission
P.O.Box 2340, D-76125 Karlsruhe, Germany
E-mail: Claudie-Kate.Black@ec.europa.eu, Evelyn.Zuleger@ ec.europa.eu

***Abstract:***

The ITU in Karlsruhe routinely analyses a multitude of samples from a wide range of internal and external customers for safeguards, often in widely different matrices. High through-put analysis techniques are employed using meticulous care to ensure accurate, precise and timely results are provided.

Thermal ionisation mass spectrometry (TIMS) is used for isotopic analysis of Uranium to determine abundance and concentration information. At ITU we employ the modified total evaporation (MTE) methodology [1] for minor isotope analysis of the U234 and U236 isotopes and have observed varying run quality between different samples. As a consequence an investigation was instigated to identify possible causes for the run differences and find a general solution for a more consistent analysis approach.

The investigation was focused around the form of Uranium (nitrate, carbonate, etc.), and after individual chemical processing, how the sample dried on the filament for analysis. The visual appearance of the samples varied significantly. Comparison of the observations with data from the analysis run indicated that impurities within the sample matrices were causing run differences due to the large amount of sample required (5 µg) for the MTE analysis.

This presentation will provide some insight into the investigations carried out and how the analysis can be improved for consistent quality routine analysis through the preparation method.

**Keywords:** TIMS; Uranium; modified total evaporation; safeguards; electrodeposition

## 1. Introduction

The MTE methodology is routinely used in laboratories around the world [1]. At the ITU in Karlsruhe this method of analysis is offered to customers in order to provide high accuracy measurements on samples with low abundance uranium 236. During the past 2 years of utilising this methodology we have experienced on several occasions difficulties in running some of the samples that have been received.
An investigation into the reasons behind why some samples were not successful in running was initialised. The hypothesis was that some forms of uranium were better suited to analysis using TIMS than other forms. The results are described in detail in this paper.

In order to eliminate the difficulties that some samples were exhibiting it was decided to investigate the use of filament electrodeposition as a loading technique to reduce or eliminate matrix input.
Filament electrodepostion is not a new technique to TIMS. It has been used routinely for many years at Los Alamos National Laboratory for the nanogram analysis of uranium [2].
For larger sample loading one has to look towards electrodeposition of alpha planchets for analysis. There are many methods of electrodeposition dependent on requirements [3]. For filament electrodeposition in a glovebox environment a technique is required that can consistently electroplate microgram amounts of

metal on a small area without producing aggressive acidic gasses. For this reason electrodeposition using buffered ammonium oxalate [4] was considered as an electrolyte for this work.

## 2. Experimental

### 2.1. Instrumentation

A Triton thermal ionisation mass spectrometer manufactured by Thermo Scientific (Bremen, Germany) was used for all the MTE measurements. This instrument is equipped with eight Faraday cup detectors and a secondary electron multiplier (SEM) detector. Two of the amplifiers are equipped with 12Ω resistors that are used with the Faraday cups chosen to measure the U234 and U236 isotopes (when not measured on the SEM). Rhenium-Rhenium double filaments were used for all measurements.

### 2.2. Materials and Solutions

IRMM 184 (IRMM, Geel, Belgium) was used as the standard for all experiments. The samples tested were of natural uranium composition. All solid samples and standards not in uranyl nitrate form were prepared by dissolution in 8M nitric acid and adjusted with water to a concentration of 40mg/g in 4M nitric acid. The exception was for the $UF_4$ salt that was dissolved using aqua regia and adjusted to 40mg/g. All samples were then prepared to a concentration of 5mg/g in a 1M nitric acid matrix for analysis by MTE.

### 2.3. Filament Loading

Rhenium double filaments (Thermo Scientific, Bremen, Germany) were placed onto an in-house manufactured filament drying instrument. One microlitre (equivalent to 5µg of Uranium) of the prepared sample/standard was placed onto the filament and dried at 0.6 Amps. When completely dried, the filaments were turned up to 2 Amps for approximately 5 seconds to oxidise the sample.

### 2.3. Filament Electrodeposition

For electrodepostion of the sample onto the filament, the filament was used as the cathode and a 0.2mm diameter platinum wire was used as the anode. The cell volume was 2.5ml.
The electrolyte chosen was 0.2M ammonium oxalate buffered to a pH of 2 with nitric acid. These were chosen as being compatible for working under glovebox conditions. This is because the major breakdown gas from ammonium oxalate under electrolysis is carbon dioxide.

## 3. Measurement Procedures

All samples and standards were measured and calculated using the MTE methodology as described in reference [1]. The SEM detector was used for the measurement of the uranium 236 isotope. For all filament dried samples, three measurements were performed bracketed with at least two standards.
Electrodeposited filaments were mainly standards and hence calculated as standards and compared with dried filament standards on the same turret. Electrodeposited samples were calculated using a mixture of dried and electrodeposited standards from the same turret.

## 4. Results and Discussion

### 4.1. Uranium Matrix Form

Observations of the dried oxidised samples on the filaments displayed a range of properties. Due to the large sample loading (5µg) required for the MTE method all samples could be clearly seen on the filament. Some dried as a thin crystal like layer, some as a crusty salt deposit and the liquid uranyl sulphate sample as if it was still in the original liquid droplet with the height of the sample approximately 1mm. Also observed, while loading and preparing the turret, was the complete detachment of some of the very crusty samples from the filaments, leading to a failed sample run. It was felt that this accounted for a large proportion of the sample fails as the sample could detach from the filament at any point during the preparation, often without the analysts knowledge.
The sum of U238 volts for each sample measurement was used as a gauge for how effectively the sample had run for the measurement. The average of the three samples measured was taken for each sample type and the results can be seen in Figure 1.

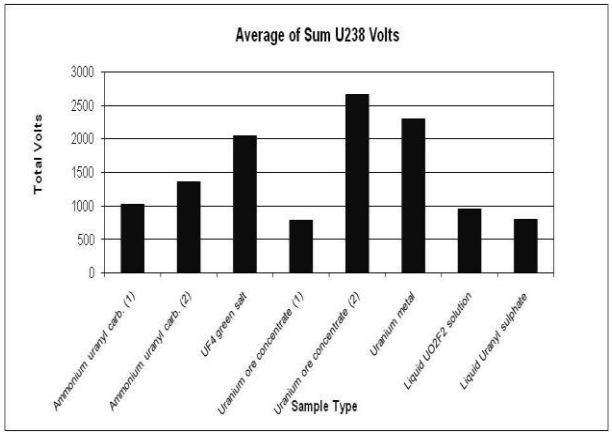


**Figure 1:** Graph showing ion yields from each type of sample analysed using MTE.

Initially it was considered that the original form of the uranium could be a significant factor in the run efficiency of the samples. However, this was found to be inconclusive from these test runs due to the differences observed between samples of the same type. In particular the largest difference observed was between two samples of uranium ore concentrate. For these two samples black particulate matter was observed in the final sample solution. Therefore it was considered that the sample matrix and other impurities that the uranium was in were more likely the determining factor in the effectiveness of the MTE run.

### 4.2. Filament Electrodeposition

In order to eliminate matrix salts and impurities affecting the filament loading process, the technique of filament electrodeposition was evaluated. Initial experiments using a small scale cell, as performed by Rokop et al. [2] for nanogram quantities of uranium, did not yield sufficient quantities of uranium on the filament for a MTE run. Increasing the volume of the cell enabled microgram quantities of uranium to be deposited onto the filament for MTE analysis.
The parameters of electrolyte pH, time and electrolysis current have been investigated but are still not fully optimised for deposition of microgram quantities of uranium. At present an electrolyte pH of approximately 2, a current of 0.1 Amps and an electrolysis time of one hour provides a sufficient quantity of uranium for analysis. However, further work is required for these parameters and others such as cathode to anode distance and cell volume to achieve optimum conditions.
Initial MTE measurements have shown that there are no differences in the uranium ratios of the IRMM 184 standard between dried filament loading and electrodeposited loading. For this standard the dried loading is still more efficient than the electrodeposited (ED) loading.

| IRMM 184 | 234/238 | 235/238 | 236/238 | Sum U238 Volts |
|---|---|---|---|---|
| Dried | 5.313E-05 | 0.007263 | 1.43E-07 | 322 |
| ED | 5.315E-05 | 0.007263 | 1.44E-07 | 689 |

**Table 1:** Comparison of average dried filament runs vs. average electrodeposition filament runs for the IRMM 184 standard, measured on the same turret.

However, fresh solutions of the uranium ore concentrate samples were prepared and run on the same turret using both loading techniques. Whilst the dried samples indicated an order of magnitude more sample on the filament in comparison to the electrodeposited samples, Table 2, it was also indicated that the ED samples plated onto the filaments in a similar manner allowing them to run more consistently.

| | Sum U238 volts | |
|---|---|---|
| Sample | Dried | ED |
| U ore con. (1) | 2624 | 289 |
| U ore con. (2) | 2078 | 282 |

**Table 2:** Comparison of dried filament runs vs. electrodeposition filament runs for the uranium ore concentrate sample type.

Due to inconsistency in drying the sample onto the filament this was the second attempt to run the dried sample filaments as the first batch of dried filaments had failed. Thus this indicates electrodepostion can eliminate differences due to matrix impurities.

## 5. Conclusions.

In conclusion, inconsistency observed between different samples is possibly due to matrix impurities within the samples or matrix effects on the drying process. Loading of these impurities with the samples can lead to increased MTE run fails, partially due to samples detaching from the filaments during the preparation and partly due to inhibition of the uranium ionisation process. Electrodeposition is indicated as a possible alternative to dry loaded samples to avoid complications from matrix impurities and increase successful filament runs in a high through/put laboratory. However, further work is

required to optimise this technique for microgram uranium loading quantities.

## 6. Acknowledgements

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Also acknowledged is Mr Adrian Nicholl for his continued technical support.

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# Age determination of plutonium for nuclear forensics

**Monika Sturm[1)], Stephan Richter[2)], Yetunde Aregbe[2)], Roger Wellum[2)], Timotheos Altzitzoglou[2)], Viktor Jobbagy[2)], Sebastién Mialle[2)], Klaus Mayer[3)], Thomas Prohaska[1)]**

1) University of Natural Resources and Life Sciences, Vienna, Austria
2) European Commission, Joint Research Centre, Institute for Reference Materials and Measurements, Geel, Belgium
3) European Commission, Joint Research Centre, Institute Transuranium Elements, Karlsruhe, Germany
E-mail: monika.sturm@boku.ac.at, stephan.richter@ec.europa.eu

***Abstract:***

*Nuclear forensics focuses on the origin and the history of seized nuclear material in order to discover its latest legal owner and production place. The chemical composition as well as the physical appearance provide information on the production process and the intended use. The "age" of the material helps to limit the number of possible facilities to those, which were active at the time of the production or processing of the material.*
*The "age" of a nuclear material is the time elapsed since the last chemical separation of mother and daughter nuclides during enrichment or reprocessing. Different "clocks" (i.e. parent/daughter nuclide pairs) can be used for the determination of the age of plutonium: $^{241}Am/^{241}Pu$, $^{238}Pu/^{234}U$, $^{239}Pu/^{235}U$ and $^{240}Pu/^{236}U$.*

*At present there are no nuclear reference materials available that are certified for the separation date of plutonium providing a solid metrological basis for age dating measurements and thus increasing the confidence in the conclusions based on these measurements. In order to meet the need of laboratories involved in nuclear forensics and age dating of nuclear material, the Institute for Reference Materials and Measurements (EC-JRC-IRMM) is closely cooperating with the Institute for Transuranium Elements (EC-JRC-ITU).*

*The widely distributed isotope ratio reference material SRM 946 (NBL CRM 136) has been dated among other samples as part of a feasibility study on plutonium reference materials certified for the separation date.*

*The results of the $^{238}Pu/^{234}U$, $^{239}Pu/^{235}$and $^{240}Pu/^{236}U$ clocks measured by isotope dilution thermal ionization mass spectrometry will be presented and discussed. The findings of the age determination agree well with the reported dates for the production of NBS 946 in 1970.*

**Keywords:** plutonium, nuclear forensics, dating, age, reference material

## 1. Introduction

Since the early 1990s more than one hundred cases of illicit trafficking of uranium and plutonium materials have been reported. These occurrences gave rise to the development of the discipline of nuclear forensic science [1]. In such cases nuclear forensic investigations are applied in addition to traditional forensics. Traditional forensics focuses on evidence such as traces (fingerprints, fibres and other residues) left behind by the culprit in order to identify linkages between (suspect) individuals and events (crimes). In contrast to that, nuclear forensics aims at re-establishing the history of the seized material and provide hints on the possible origin of the material. To this end,

the isotopic and elemental composition (major elements and impurities) in combination with dimensions of larger objects, and particle form and particle size of powders, respectively, are investigated. These findings help in determining the production process and/or the purpose the material was intended for by comparison with the entries in databases containing facility specific attributes of nuclear material. Another key parameter helping to discover the latest legal owner and production place of seized nuclear material is the “age” of the material.

## 1.1. Principles of plutonium age dating

The time span since the last removal of the “daughter” nuclides (decay products) from the “mother” nuclides is defined as the "age" of nuclear material such as plutonium. Such manipulations that separate the mother nuclides from their daughter nuclides - and thus set the “clock” to zero - occur for instance during reprocessing. The following isotopic systems are commonly used as "clocks" for the age dating of plutonium: $^{241}Pu/^{241}Am$, $^{238}Pu/^{234}U$, $^{239}Pu/^{235}U$ and $^{240}Pu/^{236}U$ [2]. Radiochronometry, however, relies on a few basic assumptions: the daughter nuclides should be completely removed from their respective parent nuclides at $t$=0, the material shall be considered as closed system, i.e. no parent or daughter nuclide shall be added or removed during the time interval under consideration, and parent and daughter nuclide shall be in radioactive disequilibrium.

The determination of the age is based on the equations describing the radioactive decay, as shown in equation 1 to 3 for the chronometer $^{239}Pu/^{235}U$ [3]. The amount of the mother nuclide $^{239}Pu$ at the time of the measurement (present time) is represented by $^{239}Pu_t$. The amount of $^{239}Pu$ at the time of the separation ($t$=0) is $^{239}Pu_0$. $^{235}U_t$ is the amount of $^{235}U$ at the present time. The decay constants $\lambda$ of mother and daughter nuclide can be calculated from their half-lives $t_{1/2}$ according to equation 3.

$$^{239}Pu_t = {}^{239}Pu_0 * e^{-\lambda_{239Pu} * t} \quad (1)$$

$$^{235}U_t = \lambda_{239Pu} * (\lambda_{235U} - \lambda_{239Pu})^{-1} * {}^{239}Pu_0 * (e^{-\lambda_{239Pu} * t} - e^{-\lambda_{235U} * t}) \quad (2)$$

$$\lambda = t_{1/2}{}^{-1} * \ln 2 \quad (3)$$

Consequently, the time span $t$ that elapsed since the last separation can be determined by calculating back to the time when no daughter nuclides were present. Assuming the complete absence of the daughter nuclides at the time of the last separation, one can thus determine when the last chemical separation of the daughter nuclides took place. Table 1 lists the parent/daughter nuclide pairs that can be applied for the determination of the age of plutonium materials.

| Clock | Half-life mother (a) | Reference | Half-life daughter (a) | Reference |
|---|---|---|---|---|
| $^{238}Pu/^{234}U$ | 87.74(3) | [4] | $2.455(6)\cdot10^{5}$ | [4] |
| $^{239}Pu/^{235}U$ | $2.410(3)\cdot10^{4}$ | [4] | $7.04(1)\cdot10^{8}$ | [4] |
| $^{240}Pu/^{236}U$ | 6561(7) | [4] | $2.342(4)\cdot10^{7}$ | [4] |
| $^{241}Pu/^{241}Am$ | 14.325(24) | [5] | 432.6(6) | [4] |
| $^{242}Pu/^{238}U$ | $3.73(3)\cdot10^{5}$ | [4] | $4.468(5)\cdot10^{9}$ | [4] |

**Table 1:** Isotope pairs formed by plutonium mother nuclides and their respective daughter nuclides

It is advisable to use more than one chronometer nuclide pair for the age dating of plutonium in order to enhance the reliability of the results and to gain additional information about the material. Using only one clock might lead to erroneous conclusions, as systematic errors (e.g. arising from incomplete separation at $t$=0) remain undetected. Inconsistent age dating results of several parent/daughter pairs may point at incomplete removal of the decay products from the parent nuclides. In case the sample consists of a mixture of two or more different materials a similar effect can occur [6]. Any addition of daughter isotopes during the time between the initial separation and the age dating measurement may also lead to inconsistent results from different clocks. The possibility of a contamination of a plutonium

sample in the facility where it was manipulated with natural or non-natural uranium should be kept in mind.

### 1.2. CRMs for plutonium age dating

Presently, no plutonium materials are available with a metrologically sound certification for their separation date, including a stated quality of the separation from daughter nuclides. The need for Certified Reference Materials (CRM) for age dating has repeatedly been expressed by members of the nuclear safeguards and nuclear forensics communities. Such CRMs would assist in providing a more solid metrological basis for age dating measurements, serve for method validation and increase the credibility of the conclusions based on these measurements.

In order to address this challenge the EC-JRC-IRMM (European Commission, Joint Research Centre, Institute for Reference Materials and Measurements) and the EC-JRC-ITU (Institute for Transuranium Elements) are closely co-operating on the certification of uranium [7] and plutonium materials certified for their separation date.

## 2. Methods and candidate CRM materials

After the chemical separation of the plutonium, uranium and americium fractions from the sample matrix, isotope dilution mass spectrometric measurements (IDMS) are performed by Thermal Ionisation Mass Spectrometry (TIMS). All TIMS measurements were done applying the total evaporation method [8]. $^{241}$Am has also been measured by gamma-ray spectrometry and alpha spectrometry. The protocols for the chemical separation procedures isolating the elements of interest (Pu, U, Am) prior to analysis were optimized successfully. This allowed the metrological determination of nuclide amount ratios formed from the determined amount contents of the respective mother ($^{238}$Pu, $^{239}$Pu, $^{240}$Pu, $^{241}$Pu, $^{242}$Pu) and daughter ($^{234}$U, $^{235}$U, $^{236}$U, $^{238}$U and $^{241}$Am) nuclides.

The first step of this study was to select suitable plutonium materials. Firstly the plutonium isotope reference materials NBS SRM 946, NBS SRM 947 and NBS SRM 948 with reported dates of the individual production stages were selected. These samples also served as samples for method development. Secondly, plutonium materials that are not described in literature were selected. The latter do not only serve as candidate materials for CRMs for nuclear age dating but are as well suitable as samples for future inter-laboratory comparison exercises on age determination of plutonium. In total seven different materials covering an age range of almost 20 years were selected. Furthermore they also have a range of plutonium isotopic compositions that covers fuel grade, reactor grade and weapon grade plutonium (classification according to [9]).
The results of the age determination of plutonium/uranium clocks of the widely distributed isotope ratio reference material NBS SRM 946 (NBL CRM 136) will be discussed hereafter. Furthermore two possible scenarios are discussed illustrating why it may be a good idea to measure - when possible - more than one clock; not only the absolute age determined by each clock helps to unravel the history of the material but also how the clocks stand relative to each other may tell a tale of its own.

## 3. Results and Discussion

As mentioned above the widely distributed isotope ratio reference material NBS SRM 946 (NBL CRM 136) has been dated among other samples. The findings of the age determination agree well with the reported dates for the production of NBS SRM 946 in 1970 [12]. The results of the age determination can be found in Table 2 and Figure 1. One can easily see in case of this sample all three commonly used plutonium/uranium clocks ($^{238}$Pu/$^{234}$U, $^{239}$Pu/$^{235}$U and $^{240}$Pu/$^{236}$U) are in good agreement with each other. In addition to that the result for the usually not employed clock $^{242}$Pu/$^{238}$U shown in Table 2 is also not significantly different from the above mentioned clocks. The clock $^{242}$Pu/$^{238}$U is normally not considered due to the low concentrations of mother and daughter nuclides, the ubiquity of $^{238}$U and challenges when measuring $^{238}$U that lead to higher uncertainties.

| Clock | Determined "age" (a) | |
|---|---|---|
| | Value | Uncertainty |
| $^{238}Pu/^{234}U$ | 41.09 | 0.29 |
| $^{239}Pu/^{235}U$ | 41.12 | 0.12 |
| $^{240}Pu/^{236}U$ | 41.16 | 0.11 |
| $^{242}Pu/^{238}U$ | 43.4 | 4.5 |

**Table 2: Results of age determination of sample NBS 946 determined by ID TIMS calculated for 18.10.2011; All uncertainties are expanded uncertainties (k=2).**

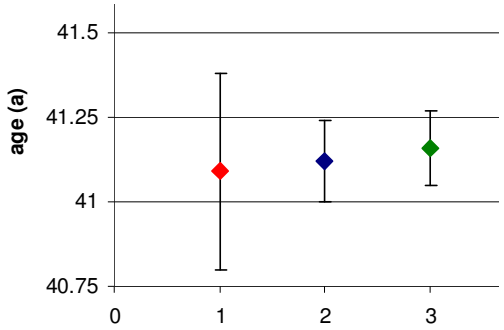


**Figure 1:** Results of age determination of sample NBS 946 determined by ID TIMS calculated for 18.10.2011 using the following clocks: $^{238}Pu/^{234}U$ (1); $^{239}Pu/^{235}U$ (2); $^{240}Pu/^{236}U$ (3); All uncertainties are expanded uncertainties (k=2).

However, the interpretation of the results of the age determination of a given sample is not always that straight forward. One possible reason for that can be that the sample under investigation is a mixture of two or more materials that were mixed; Apart from that is possible that the separation procedure did not remove the daughter nuclides completely at time zero or that a contamination of the plutonium sample with natural uranium or other uranium occurred. The latter two lead to characteristic patterns of the plutonium/uranium clocks relative to each other and can thus not only help to interpret the results in a more concise way but also may give additional information of the possible history of the sample. Care must be take, however, not to attempt such interpretations without careful consideration of effects on the age dating results of the different clocks that are artefacts of the sample preparation and measurement procedure such as incomplete separations during sample preparation or corrections one should apply in order to obtain interpretable results.

In case of a contamination of the plutonium sample with uranium it is quite likely that one will observe that the result for $^{239}Pu/^{235}U$ clock is shifted to a higher age relative to the $^{238}Pu/^{234}U$ and $^{240}Pu/^{236}U$ dating results. Apart from that the $^{242}Pu/^{238}U$ age dating result will rise to a very high value which is a clear sign for a contamination with natural uranium.

Another scenario that shifts the plutonium/uranium clocks relative to each other is incomplete removal of the daughter nuclides during the last manipulation of the material. In that case the results of all plutonium/uranium clocks will be shifted to an older age due to the excess of uranium. However, the result for the $^{238}Pu/^{234}U$ clock is shifted to a younger age relative to the $^{239}Pu/^{235}U$ and $^{240}Pu/^{236}U$ dating results. This can be explained by the half lives of the respective mother daughter nuclide pairs shown in Table 1.

## 4. Conclusions and Outlook

The age determination of nuclear material is challenging and not always straight forward due to the nature of the samples; thus for the age estimation of a sample, such as material seized in nuclear forensics investigations, it is advisable to use more than one clock. This helps to ensure the reliability of the results and to exclude the possibility that the sample under investigation is for instance a

mixture of two or more materials, has not been separated completely at time zero or has been contaminated with for instance uranium. The latter two lead to characteristic patterns of the plutonium/uranium clocks relative to each other and can thus not only help to interpret the results in a more concise way but also may give additional information of the possible history of the sample.

To meet the needs of the nuclear safeguards and security community, the EC-JRC-IRMM and the EC-JRC-ITU are closely cooperating towards the production of nuclear CRMs certified for the separation date suitable for nuclear age dating.

## 5. Acknowledgements

The authors would like to thank Rüdiger Kessel for providing for the program GUM work bench 2.4 (Metrodata, Germany) for the uncertainty evaluation.

## 6. Legal matters

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# Determination of Accurate Gamma-Ray Signatures for $^{233}$U

**L.G. Evans, R.D. McElroy, A.M. Krichinsky, S. Croft, S.L. Cleveland**

Safeguards & Security Technology Group,
Nuclear Security and Isotope Technology Division,
Oak Ridge National Laboratory, P.O. Box 2008, Oak Ridge, TN 37831, USA
E-mail: evanslg@ornl.gov

***Abstract:***

*Uranium-233 is a fissile, weapons-useable material, and needs to be appropriately safeguarded. The accurate identification and quantification of $^{233}$U using non-destructive assay (NDA) methods are fundamental aspects of nuclear safeguards applied to a civilian thorium fuel cycle or to the detection of clandestine fuel cycle activities (i.e., breeding fissile material). Currently, relatively large uncertainties associated with existing $^{233}$U gamma-ray decay data lead to commensurately large uncertainties in determining signatures for $^{233}$U by gamma-ray spectrometry. This introduces unreliability when verifying the presence and quantity of $^{233}$U as part of nuclear safeguards, nonproliferation, and forensic activities. Improvements in $^{233}$U nuclear data will positively impact the work of technicians taking inventories, inspectors monitoring plants, and researchers developing new detection technologies.*

*The Safeguards & Security Technology Group at Oak Ridge National Laboratory (ORNL) has a singular opportunity to access and investigate gamma-ray signatures for ultra-pure $^{233}$U via a program of work focused on preserving these materials which, otherwise, would have faced down blend (with depleted uranium) and disposal. Such pure materials (all are >99% $^{233}$U) are candidates to be used as certified reference materials (CRM) in isotope dilution mass spectrometry – the most precise method for analyzing uranium isotopic content and concentration, especially for trace samples. These pure $^{233}$U materials have low $^{232}$U content [<1 ppb in the purest item (for a 99.9986%-pure $^{233}$U item) and <1 ppm for most other items being preserved], which presents a significant advantage to high-accuracy gamma-ray measurements because it dramatically reduces the interfering gamma-ray flux from $^{232}$U and its progeny, most notably $^{208}$Tl. Lower $^{232}$U content enables a cleaner measured $^{233}$U spectrum, resulting in more accurate $^{233}$U yield determinations for the many photopeaks available from the decay of $^{233}$U and its progeny.*

*This paper reports on the first gamma-ray spectra obtained from these ultra-pure $^{233}$U items, acquired using high-resolution gamma-ray spectroscopy.*

**Keywords:** uranium-233; safeguards; gamma-ray; signatures; nuclear data

## 1. Introduction

This paper describes developments in the gamma-ray spectroscopy portion of a research effort to determine accurate gamma-ray and neutron signatures for uranium-233 ($^{233}$U). This work is a component of a larger U.S. program to preserve $^{233}$U samples with high isotopic purity (>99.4% $^{233}$U) [1]. This program is sponsored by the Next Generation Safeguards Initiative (NGSI) of the U.S. Department of Energy (DOE) National Nuclear Security Administration (NNSA) Office of Nonproliferation and International Security (NIS).

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There are two main objectives of the $^{233}$U signatures research project:

- Acquire gamma-ray and neutron signatures from ultra-high-purity $^{233}$U samples prepared as part of the NGSI ultra-high-purity $^{233}$U preservation effort.
- Obtain more accurate gamma-ray absolute emission probability ("branching ratio") data for the ultra-high-purity samples using germanium-based, high-resolution gamma-ray spectroscopy (HRGS) to improve the existing nuclear data sets available. The goal is to achieve <1% relative uncertainty in the absolute emission probability, so that the results are beneficial to evaluated nuclear data efforts.

This paper focuses on gamma-ray NDA signatures.

## 1.1. Safeguarding $^{233}$U

Uranium-233 is a man-made, fissile isotope of uranium. Pure $^{233}$U is a direct-use (weapons-useable) material [2, 3]. Under International Atomic Energy Agency (IAEA) safeguards protocols, pure unirradiated $^{233}$U is treated identically to unirradiated plutonium (containing less than 80% $^{238}$Pu), where a quantity of 2 kg or more is defined as a Category I material [4, 5]. Therefore, as an analogue of $^{239}$Pu, $^{233}$U is equally important to consider from a safeguards perspective.

Uranium-233 is produced during the neutron irradiation of fertile, natural thorium ($^{232}$Th) in a nuclear reactor. Following neutron capture, $^{233}$Th undergoes two successive $\beta^-$ decays to $^{233}$U [6]. This production mechanism is illustrated in Figure 1.

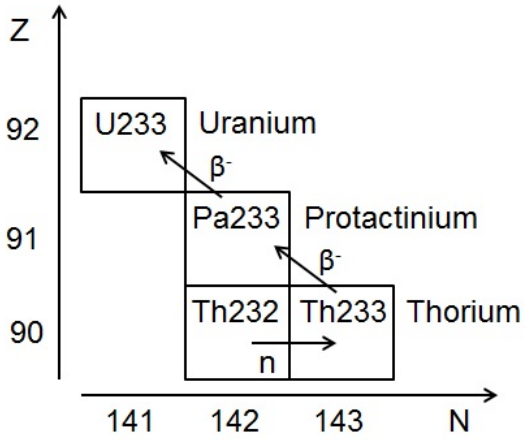


**Figure 1:** $^{233}$U production

Safeguards requirements for $^{233}$U are becoming increasingly relevant with growing interest in the use of thorium-fueled nuclear power worldwide. The ability to accurately detect, identify, and quantify $^{233}$U using non-destructive assay (NDA) methods are fundamental requirements of a nuclear safeguards program applied to a civilian thorium fuel cycle or to the detection of clandestine fuel cycle activities (e.g., breeding fissile material).

## 1.2. $^{233}$U signatures

A nuclear safeguards inspectorate must be able to draw credible conclusions regarding declared or undeclared fuel cycle activities from the quantities (e.g., mass, enrichment) of special nuclear material (SNM) measured by NDA methods. Where nuclear data are limited or where results are analyzed with nuclear data of poor accuracy, greater uncertainties are introduced in the final assay result. This reduces the overall confidence level in the measured quantities upon which safeguards conclusions are based.

The successful identification and quantification of $^{233}$U using NDA methods is hampered by the poor quality and limited availability of existing $^{233}$U nuclear data. For example, the quantitative assay of $^{233}$U-bearing items using germanium-based HRGS requires accurate, known values of the gamma-ray

energies and absolute emission probabilities for all isotopes within the measured item. A literature review [6–10] reveals that the absolute emission probabilities for a majority of characteristic gamma-rays from $^{233}$U have uncertainty of 10–15% or greater. In the worst cases, the absolute emission probabilities have uncertainty of approximately 43%. The uncertainties in the absolute emission probabilities are displayed as a function of gamma-ray energy in Figure 2 to illustrate this problem.

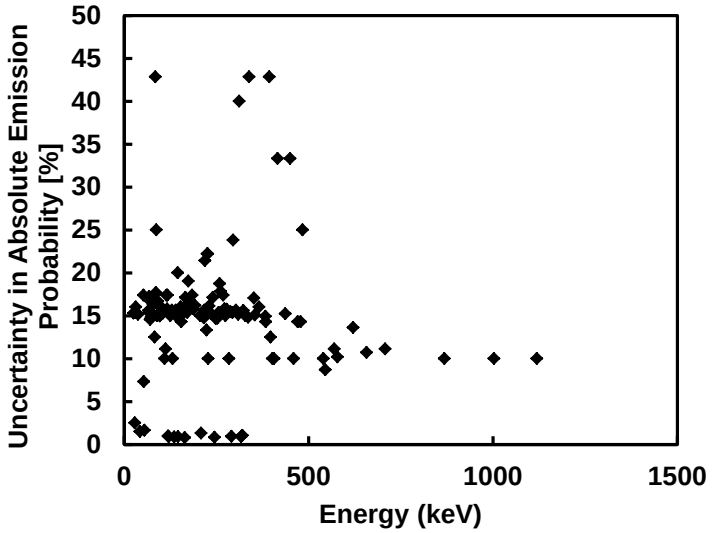


**Figure 2:** Literature review data from various sources [6-10] plotted to show the uncertainty in the gamma-ray absolute emission probability as a function of energy.

It could be expected that this uncertainty in absolute emission probability would trend *inversely* with the magnitude of absolute emission probability itself, in other words, with the least intense lines being harder to determine (subject to lower counting statistics) and, therefore, more uncertain. However, it is surprising that this is not always the case. For example, the $^{233}$U line at 120.816 (± 0.001) keV [7] has an absolute emission probability of 0.00332 (± 0.00003) [7] i.e., a relative uncertainty of 0.9%. A line of similar energy and intensity has a very different relative uncertainty; the $^{233}$U line at 217.159 (± 0.002) keV [8] has an absolute emission probability of 0.0032 (± 0.0005) [8] i.e., a relative uncertainty of 15.6%. The reason for this and similar apparent anomalies in the $^{233}$U nuclear data must be identified.

Furthermore, out of the ~130 documented lines for the $^{233}$U decay scheme [7-11], the IAEA has adopted data for the 11 lines with an uncertainty of 1.6% or less in a recent report of data recommendations for safeguards [7]. However, these 11 reported lines all have energies of <321 keV. This is an important observation for the safeguards assay of shielded $^{233}$U-bearing materials, where low-energy gamma-ray attenuation would reduce the observed intensity of these lines. This is especially important for items with elevated $^{232}$U content because there are a high number of interfering lines, together with a strong Compton continuum, in this low-to-mid energy region. In these two scenarios (i.e., shielding and elevated $^{232}$U content), high-energy lines must be relied on for the determination of isotopic content; but these are the lines with higher uncertainty in the absolute emission probability. These safeguards application examples demonstrate the need to revisit, re-evaluate, and, where possible, improve nuclear data for $^{233}$U.

The collection of accurate gamma-ray signature data from $^{233}$U will be used to improve the existing nuclear data sets available. Updating the safeguards database for the assay of $^{233}$U will improve the accuracy of gamma-ray NDA methods and, therefore, reduce the uncertainty in results obtained by NDA. In turn, this will enable an inspectorate to draw conclusions with regard to declared or undeclared activities with greater confidence to help achieve the aim of improving the early detection probability for undeclared fuel cycle activities.

## 2. Signature acquisition

Ultra-high-purity $^{233}$U has a role in improving nuclear data for safeguards gamma-ray NDA measurements. Access to ultra-high-purity $^{233}$U samples (i.e., isotopic purity >99% $^{233}$U) coupled with instrumentation for performing HRGS will enable gamma-ray signature acquisition and a re-evaluation of the $^{233}$U gamma-ray absolute emission probabilities. Our goal is to measure absolute emission probabilities to better than 1% relative uncertainty across a broad range of gamma-ray energies – focusing on the 600-to-800 keV energy region.

### 2.1. High-isotopic-purity $^{233}$U

Oak Ridge National Laboratory (ORNL) hosts the U.S. national repository for separated $^{233}$U (i.e., $^{233}$U that has been separated from thorium target material and fission products) in the oldest operating nuclear facility in the world [12]. Separated $^{233}$U is stored in vaults within this facility. This material includes some of the world's purest $^{233}$U with isotopic purity >99% $^{233}$U [4], and some samples available at greater than four 9's purity (>99.99%). Under the NGSI-sponsored high-purity $^{233}$U preservation effort, this material is being rescued and preserved to provide feed for certified reference materials (CRM). CRM serves the scientific, safeguards, nonproliferation, and forensics communities for high precision uranium analyses – notably, isotope dilution mass spectrometry (IDMS), the most precise trace-uranium analytical technique. This preservation effort ensures that the value of the high-isotopic-purity $^{233}$U items is not permanently lost to pending disposition activities. The research presented here is timely because the preservation effort offers a singular opportunity to access this unique material for high-precision NDA measurements.

A measurement campaign is taking place to acquire the gamma-ray spectra from each of the ultra-high-purity $^{233}$U items following retrieval from their current storage location but prior to shipment off-site. NDA measurements are taking place in parallel with other $^{233}$U preservation activities to minimize project costs and to eliminate the need to handle the material multiple times.

Ultra-high-purity $^{233}$U items that are being recovered and made available for measurement come in a variety of chemical and physical forms. The shapes of the $^{233}$U materials and packages vary widely, in the same way that items under assay in a facility may be expected to vary. Figure 3 shows one of the samples that have been unpacked and measured. Gamma-ray spectra acquired on the items as presented lend themselves to creating a useful archive for subsequent analysis to extract attributes of safeguards interest, in a manner similar to in-field analysis.


**Figure 3:** Item BA-35-1. Radiograph from an inspection at the repository (left). Welded capsule removed from the outer can (right).

Item BA-35-1, shown in Figure 3, contains 4g of $^{233}$U metal as a foil 2.625 inches (66.675 mm) in diameter with a 0.0035 inch (0.0889 mm) thickness – so thin that it is almost invisible in the radiograph. The $^{232}$U content of the item is 1 ppm. It is packed in a double-layer of welded, stainless steel capsules, the outer layer of which is 1.25 inches (31.75 mm) in diameter and 4.5 inches (114.3 mm) in height. This item was the first of the ultra-high-purity $^{233}$U items to be measured as part of this $^{233}$U signatures project.

## 2.2. Instrumentation

A single, high-resolution, Broad Energy Germanium (BEGe) detector (Canberra model BE2820 with 59.71 mm crystal diameter x 20 mm crystal thickness) is used for gamma-ray detection to facilitate simultaneous measurement of the low- and high-energy portions of the gamma-ray spectrum. The high-resolution gamma-ray detector was chosen to resolve the complex spectrum for $^{233}$U, which has many lines that are close together in energy. A commercially available, copper-lined, shield assembly and custom-fabricated collimator set are used to reduce the room background contribution to the detected spectrum and, therefore, reduce background interferences. This low background condition helps to identify the low-intensity $^{233}$U peaks. The collimator/shield assembly is shown in Figure 5. Collimation also served to reduce the detector dead time in response to these high-activity samples and therefore reduce the counting losses, which could present a problem when recording counts from the low-intensity $^{233}$U lines.

The measurement times are long in duration (typically >24 h) to obtain sufficient counting statistics from the-low intensity $^{233}$U lines, and to identify and remove the interferences from the $^{232}$U impurity and daughters. This is necessary to achieve the high-quality, basic nuclear data sought. Applied measurements would not need such long counts.

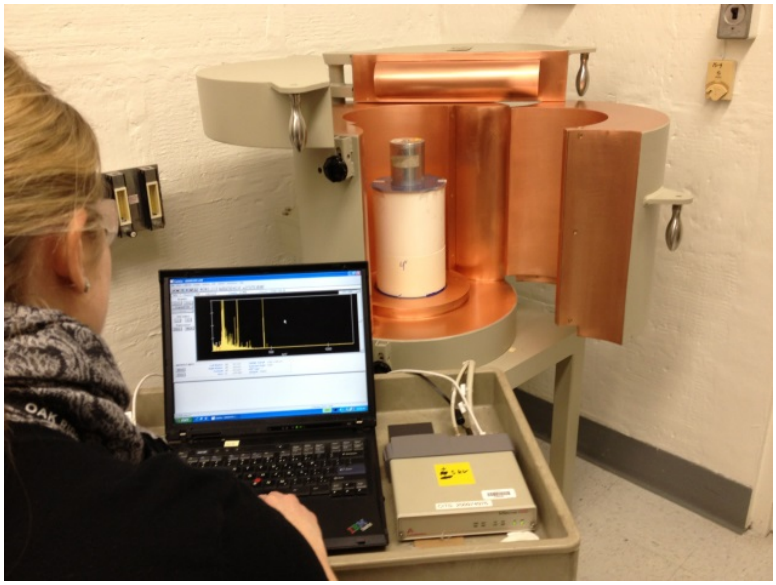

**Figure 5:** Gamma-ray collimator/shield assembly and measurement configuration.

The Canberra Genie™ 2000 Gamma Acquisition & Analysis software package (V3.2.1, 2009) [13] was chosen to control the detector high voltage and multi-channel analyzer (MCA) data acquisition, and also for the initial analysis of the acquired gamma-ray spectra.

# 3. $^{233}$U Gamma-ray signatures

## 3.1. Gamma-ray origins

Uranium-233 material contains $^{232}$U and $^{233}$U isotopes together with daughter nuclides for the decay series of both isotopes [14]. The $^{233}$U decay series is shown in Figure 6, and the $^{232}$U decay series is shown in Figure 7 [10, 15, 16]. The $^{229}$Th daughter can provide an estimate of the time since purifying the material using the $^{233}$U/$^{229}$Th mass ratio [14]. The high $^{233}$U purity (i.e., low $^{232}$U content) significantly reduces the interferences normally encountered in the assay of $^{233}$U. Comparisons with less pure materials will facilitate the separation of lines generally associated with $^{232}$U. Measurements before and after chemical separation are also planned.

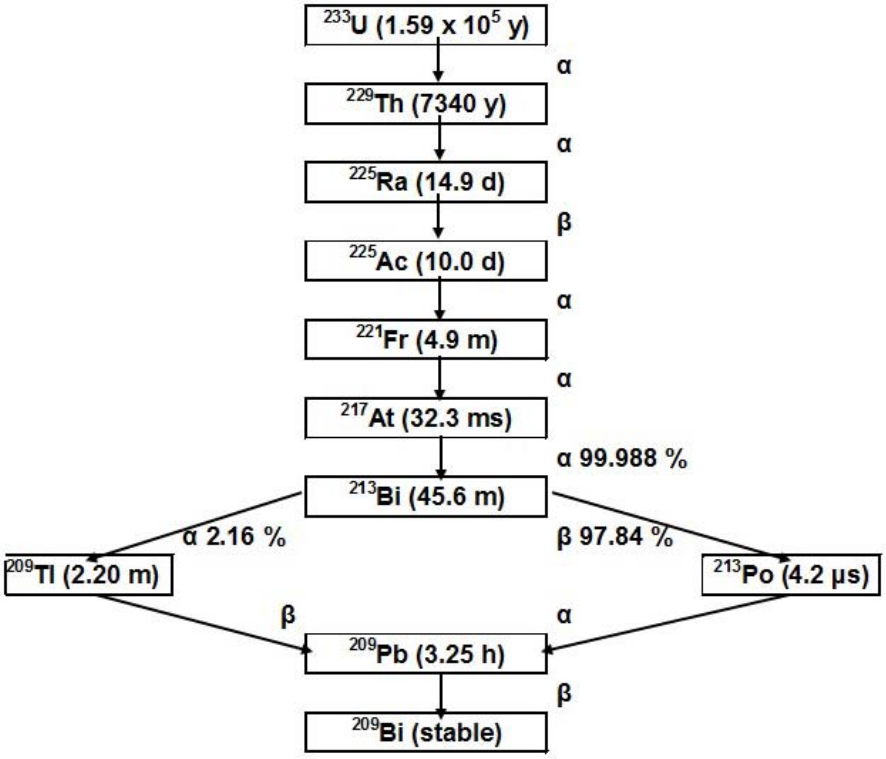


**Figure 6:** $^{233}$U decay series.

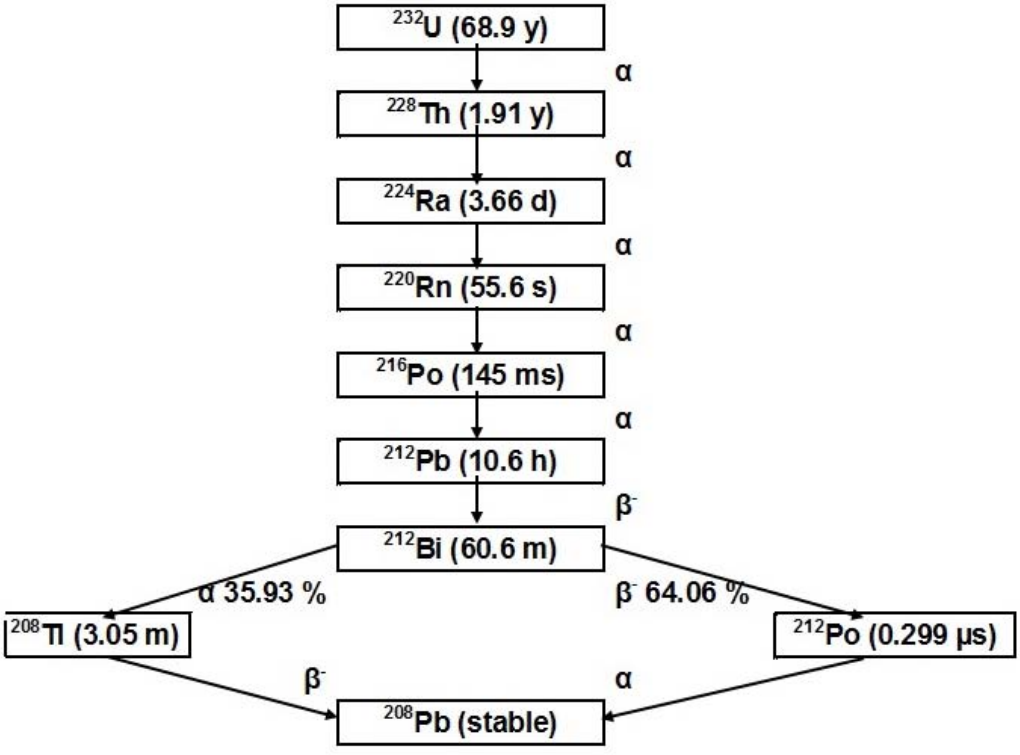


**Figure 7:** $^{232}$U decay series.

## 3.2. Preliminary measurements

One of the first $^{233}$U gamma-ray spectra was acquired over a 48 h period at a low gain setting to maximize the energy region of interest (Figure 8). These data were acquired in the range 0.1 keV to 4799 keV, binned in 16K channels. The overall shape of the spectrum is characterized by a high Compton background continuum due to a significant contribution from the high-energy gamma-rays emitted from this material and thus partial photon energy loss in the detector crystal. A custom library was created for the $^{233}$U decay series because such a library was not available in the Genie 2000 standard library files. Peaks are labeled based on the library-directed search using this new library.

There is no visible presence of a gamma-ray peak at 661.6 keV, which indicates that the $^{137}$Cs fission product has been removed thoroughly from this material. Therefore, it can be concluded that this is clean, separated material, and it has not been re-irradiated.

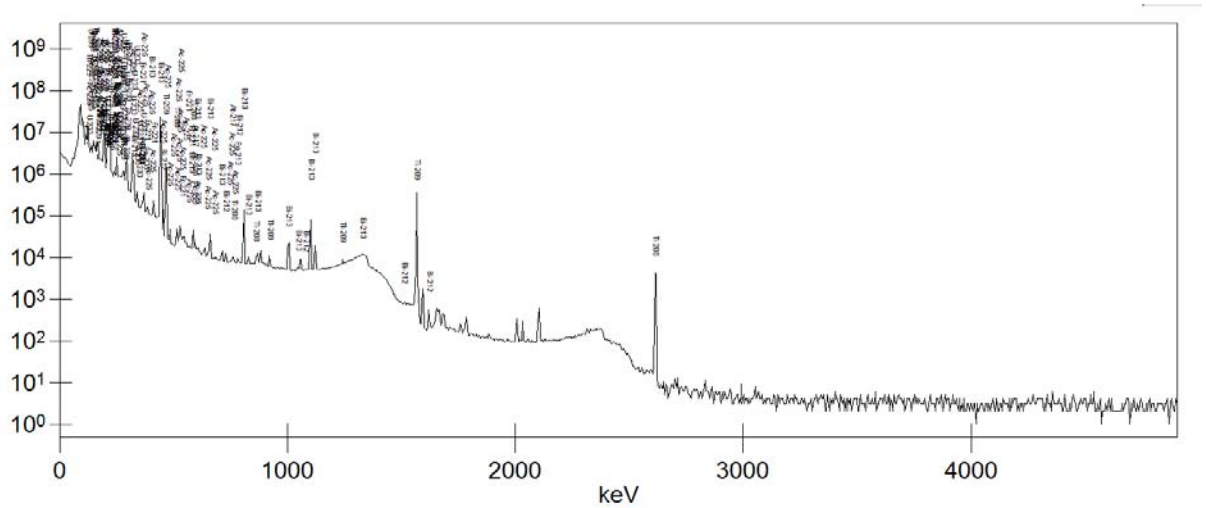


**Figure 8:** Gamma-ray spectrum (counts per channel) from item BA-35-1 $^{233}$U foil.

Gamma-ray regions of significance for the relative isotopic determination and mass quantification of $^{233}$U content have been identified by Harker and Sampson [15]. They identify three distinct peaks due to gamma-rays emitted directly from $^{233}$U at 317.2 keV, 320.5 keV, and 323.4 keV, with the third peak being a triplet of separate contributions from 323.4 keV $^{233}$U, 323.7 keV $^{213}$Bi, and 324.1 keV $^{221}$Fr. These three distinct peaks (including the triplet) are all clear indicators of $^{233}$U content and can be seen in the region of the spectrum acquired for the BA-35-1 foil shown in Figure 9. The intense 440.44 keV gamma-ray peak from $^{213}$Bi is also evident in Figure 9.

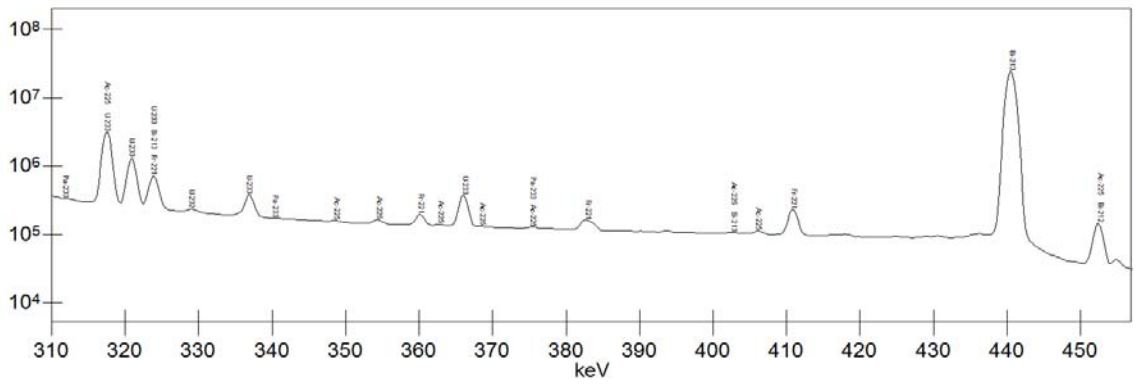


**Figure 9:** Gamma-ray spectrum (counts per channel) from item BA-35-1 $^{233}$U foil with energy region of interest from 310 keV to 455 keV.

## 4. Discussion

Measured gamma-ray nuclear data is only as accurate as the efficiency calibration and, therefore, is limited in accuracy by the efficiency curve used to generate that data. In order to better meet the primary goal of the $^{233}$U signatures project, which is to improve the available $^{233}$U gamma-ray absolute emission probability data by reducing the uncertainties in this data to 1% or less, ideally, the relative efficiency curve must be well known (to well within 1%) so as not be a dominant component of uncertainty in the final measured absolute emission probability values. To obtain an accurate estimate of the relative efficiency at this level, the internal geometry and packaging of the measured item must be well known, for accurately determining the attenuation of the gamma-rays before they exit the item packaging and for accurately determining the geometric component of detection efficiency (solid angle of the gamma-rays impinging upon the detector). The relative efficiency curve has been somewhat more challenging to determine than expected, since the internal geometries of the measured items are not well known (as demonstrated by the $^{233}$U foil rolled inside the capsule shown in Figure 3). The attenuation must be estimated and corrected by considering that groups of lines with similar energies are attenuated to the same degree [17]. When done with care and attention to detail, this method will likely reduce the uncertainties in the existing nuclear data; however, it is difficult to achieve the desired nuclear data quality results of <1% for the absolute emission intensities because uncertainties in item geometries will propagate through these correction factors. This geometric contribution to uncertainty is being addressed with counting standards made from ultra-high purity $^{233}$U, discussed in the future work section below.

## 5. Conclusions

The first measurement campaign is under way to characterize the ultra-high-purity $^{233}$U items recovered and repackaged from the repository at ORNL. The first gamma-ray spectra have been acquired and studied. A $^{233}$U nuclide library has been created following an extensive literature review of the available nuclear data. This library will augment the existing nuclide libraries within Genie 2000 and, therefore, extend its analysis capability for this work, thereby enabling the library-assisted analysis of these spectra. This is an iterative process as improved emission probabilities are substituted in to this library. More work remains to be done on generating relative efficiency curves for the measured items and completing the measurement of additional items now available from the repository.

## 6. Future Work

An ongoing portion of the $^{233}$U signatures project will take advantage of an associated related $^{229}$Th recovery effort to create small, chemically purified $^{233}$U items where the $^{233}$U daughters and, more significantly, $^{232}$U daughters have been removed. Approximately 1 mg of this purified $^{233}$U in solution will be pipetted on to each of three stainless steel planchets and flamed to an oxide, thus providing a near-ideal counting geometry, with very little self-attenuation that can be set easily at a reproducible distance from the detector end-cap. Examples of planchet geometries available for measurement are shown in Figure 10. A stand will be custom machined to hold a planchet in place above the detector during each measurement. This will simplify the efficiency calibration process as well as generate a more accurate efficiency curve. It is anticipated that measurements of this item will provide our most accurate results.

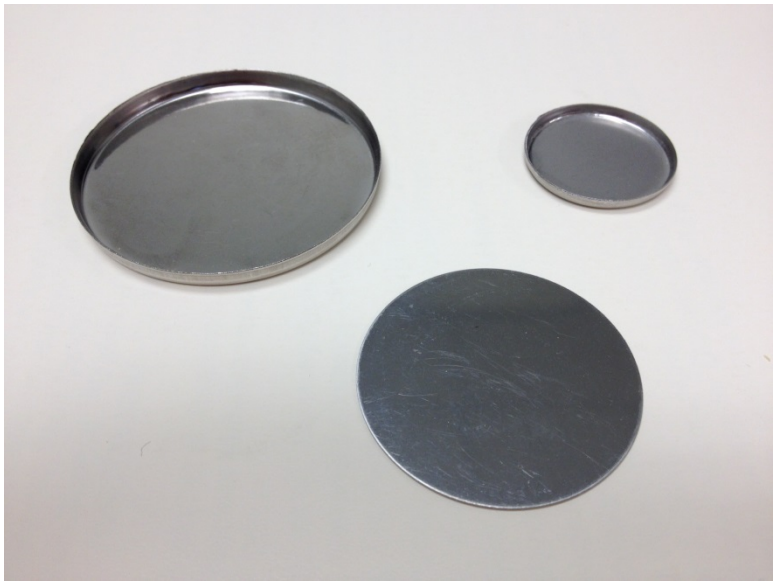

**Figure 10:** Planchet geometries available for measurement.

In addition to anomalies in nuclear data used for the characterization of gamma-ray signatures, almost no data exists for the active neutron interrogation of $^{233}$U using well-characterized reference materials. Therefore, future work will expand the $^{233}$U signatures project to neutron NDA methods. During future work, neutron measurements will also be performed on bulk quantity items in the active well neutron coincidence counter (AWCC) to generate data for safeguards neutron measurements. The AWCC is the principal neutron counter for the quantitative mass assay of dense uranium bulk materials including oxides and metals. The AWCC employs the technique of active neutron interrogation using two $^{241}$AmLi (α, n) sources to induce fission within the item being measured. Each of the ultra-high-purity $^{233}$U items will be measured in ORNL's existing AWCC. There is also an opportunity to measure these items in ORNL's $^{252}$Cf shuffler system.

## 7. Acknowledgements

The authors would like to acknowledge and thank the Next Generation Safeguards Initiative (NGSI), Office of Nonproliferation and International Security (NIS), National Nuclear Security Administration (NNSA) of the U.S. Department of Energy (DOE) for funding this safeguards technology development research project.

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# Application of the two-group - one-region and two-region - one-group Feynman-alpha formulas in safeguards and accelerator-driven systems (ADS).

**Dina Chernikova[1], Imre Pázsit[1], Wang Ziguan[1]**

[1]Chalmers University of Technology, Department of Applied Physics, Nuclear Engineering, Fysikgarden 4, SE-412 96 Gothenburg, Sweden
E-mail: dina@nephy.chalmers.se

***Abstract:***

*The applicability of the two-group (one-region) and two-region (one-group) Feynman-alpha (variance to mean) formulas was assessed with regards to applications in safeguards and accelerator-driven systems (ADS), considered as an option for transmutation of nuclear wastes. Since two-group calculations with the master equation technique when both thermal and fast fissions are included, have not been performed earlier, investigation of this problem has a methodological value of its own. The potential applications of the two-group - one-region and two-region - one-group Feynman-alpha approaches in nuclear safeguards were evaluated and compared to the results of Monte-Carlo simulations.*

**Keywords:** Feynman method; safeguards; ADS

## 1. Introduction

The main idea behind the Feynman-alpha (variance to mean) method is the measurement of neutron counts and evaluation their relative variance as a function of time. Since in nuclear fission process the number of neutrons produced per fission is not a constant value and is more than one, the ratio between the variance of counts to the expected number of counts will deviate from unity as a function of measurement time.

Originally, this property was used for the determination of criticality of a subcritical multiplying system, such as an ADS, for example. Later on, it was applied also in nuclear Safeguards in order to determine the presence of nuclear material.

In the present work we will try to evaluate the sensitivity of the Feynman-alpha approach to the evaluation of the isotopic content of nuclear fuel or waste, based on the Feynman-alpha formulas extended in the present paper.

This work has a methodological value of its own, since it is related to the fact that the traditional one-group (one-region) variance to mean formula was elaborated and used for thermal systems in which the thermal flux and the lifetime of thermal neutrons dominates. However, this approach does not fully describe fast neutron systems, as well as heavily reflected thermal systems, since the effects of the reflector play a significant role in the latter.

Thus, a two-group (or two-point) master equation approach might lend the possibility of treating fast reflected systems in a more accurate way, by treating the counts separately in the fast and the thermal groups (or in the nuclear material and reflector regions).

## 2. The main concept and equations

In our recent papers we described the derivation of the two-group (one-point) Anderson [1] and two-point (one-group) Anderson [2] version of Feynman-alpha theory by using the Kolmogorov forward approach.

However, the two-group (one-point) formulas derived in Anderson [1] were limited to the case of only fast neutron detections with also neglecting of fast fissions. In the present work,

we extended the two-group (one-point) Feynman-alpha theory by including these missing terms.

The two-point (one-group) variance-to-mean formula was also enhanced, by including detection and source terms in both regions. This extension opens the possibility to apply this version of the formulas to the case when nuclear material, itself, is the source of neutrons.

Thus, results of the derivation of the extended analytical expressions for two-point (one-group) and two-group (one-point) Feynman-alpha formulas are presented below. The full details of derivations with inclusion of delay neutrons etc. can be found in [3].

## 2.1. Two-group one-point variance-to-mean formulas

In the model, used for derivation of the extended version of the two-group (one-point) Feynman-alpha formulas, we assumed that the medium is infinite and homogeneous. The neutron population consists of two groups of neutrons: fast (denoted as 1) and thermal (denoted as 2). In the model we also include a Poisson source of fast neutrons with strength $S_1$.

Fast and thermal neutrons can undergo different reactions (*i*), i.e.:

- absorption (*i*=*a*),
- fission (*i*=*f*),
- detection (*i*=*d*),
- removal from the fast group to the thermal (*i*=*r*=*R*).

Each of these reactions for different group of neutrons can be described by transition intensities, as shown in Fig. 1.

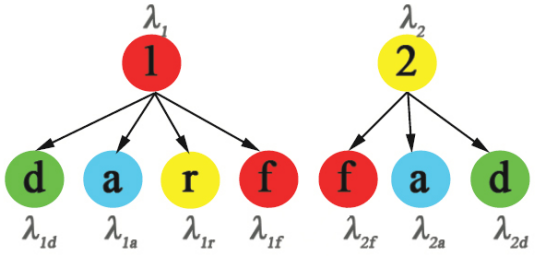


**Figure 1:** a two-group (one-point) model of various processes which particles can undergo.

Total transition intensities for fast and thermal neutrons are denoted as $\boldsymbol{\lambda_1}$ and $\boldsymbol{\lambda_2}$, respectively:

$$\begin{aligned}\lambda_1 &= \lambda_{1f} + \lambda_{1a} + \lambda_R + \lambda_{1d} \\ \lambda_2 &= \lambda_{2f} + \lambda_{2a} + \lambda_{2d}\end{aligned}$$

The final expression of two-group (one-point) Feynman-alpha formulas is given below:

$$\frac{\sigma^2_{ZZ}(t)}{\langle Z \rangle} = 1 + Y_1\left(1 - \frac{1-e^{-\omega_1 T}}{\omega_1 t}\right) + Y_2\left(1 - \frac{1-e^{-\omega_2 t}}{\omega_2 t}\right)$$

Where,

$$\begin{aligned}-\omega_1 &= \frac{1}{2}\left(-\lambda_1 - \lambda_2 + \lambda_{1f}\nu_1 - \sqrt{(-\lambda_1 + \lambda_2 + \lambda_{1f}\nu_1)^2 + 4\lambda_{2f}\lambda_R\nu_2}\right) \\ -\omega_2 &= \frac{1}{2}\left(-\lambda_1 - \lambda_2 + \lambda_{1f}\nu_1 + \sqrt{(-\lambda_1 + \lambda_2 + \lambda_{1f}\nu_1)^2 + 4\lambda_{2f}\lambda_R\nu_2}\right)\end{aligned}$$

If detection is performed with a detector of fast neutrons, such as a liquid scintillation detector, then the functions $Y_1$ and $Y_2$ should be used in the form like shown below:

$$\begin{aligned}Y_1 &= \frac{2\lambda_{1d}\left(\mu_{XX}\lambda_2 - \mu_{XX}\omega_1 + \mu_{XY}\lambda_{2f}\nu_2\right)\omega_1\omega_2}{S_1\lambda_2\omega_1\left(\omega_2 - \omega_1\right)r'} \\ Y_2 &= \frac{2\lambda_{1d}\left(\mu_{XX}\lambda_2 - \mu_{XX}\omega_2 + \mu_{XY}\lambda_{2f}\nu_2\right)\omega_1\omega_2}{S_1\lambda_2\omega_2\left(\omega_1 - \omega_2\right)r'}\end{aligned}$$

However, if a detector of thermal neutrons is used instead, e.g. $^3$He-counter, than following $Y_1$ and $Y_2$ functions must be used:

$$\begin{aligned}Y_1 &= \frac{2\lambda_{2d}\left(\mu_{YY}\lambda_1 + \mu_{XY}\lambda_R - \mu_{YY}\omega_1 + \mu_{YY}\lambda_{1f}\nu_1\right)\omega_1\omega_2}{S_1 r'\lambda_R\omega_1\left(\omega_2 - \omega_1\right)} \\ Y_2 &= \frac{2\lambda_{2d}\left(\mu_{YY}\lambda_1 + \mu_{XY}\lambda_R - \mu_{YY}\omega_2 - \mu_{YY}\lambda_{1f}\nu_1\right)\omega_1\omega_2}{S_1 r'\lambda_R\omega_2\left(\omega_1 - \omega_2\right)}\end{aligned}$$

where

$$\begin{aligned}\mu_{XX} &= \frac{\left(\omega_1\omega_2 + \lambda_2^2\right)\left(\lambda_{1f}\nu''_1\bar{N}_1 + \lambda_{2f}\nu''_2\bar{N}_2 + S_1 r''\right)}{2\left(-\lambda_{1f}\nu_1 + \lambda_1 + \lambda_2\right)\omega_1\omega_2} \\ \mu_{XY} &= \frac{\lambda_2\lambda_R\left(\lambda_{1f}\nu''_1\bar{N}_1 + \lambda_{2f}\nu''_2\bar{N}_2 + S_1 r''\right)}{2\left(-\lambda_{1f}\nu_1 + \lambda_1 + \lambda_2\right)\omega_1\omega_2} \\ \mu_{YY} &= \frac{\lambda_R^2\left(\lambda_{1f}\nu''_1\bar{N}_1 + \lambda_{2f}\nu''_2\bar{N}_2 + S_1 r''\right)}{2\left(-\lambda_{1f}\nu_1 + \lambda_1 + \lambda_2\right)\omega_1\omega_2}\end{aligned}$$

$$\begin{aligned}\bar{N}_1 &= \frac{S_1\lambda_2 r'}{\omega_1\omega_2} \\ \bar{N}_2 &= \frac{S_1\lambda_R r'}{\omega_1\omega_2}\end{aligned}$$

Here, *ν'*, *ν''*, *r'* and *r''* stand for the first and second factorial moments of the neutrons emitted in a fission process and in a source event, respectively.

## 2.2. Two-point (one-group) variance-to-mean formulas

The assumption behind the model used in derivation of two-point (one-group) Feynman-alpha theory is that two adjacent homogeneous half-space regions (denoted as A and B) with independent reaction intensities for detection ($\boldsymbol{\lambda_{Ad}}$, $\boldsymbol{\lambda_{Bd}}$), absorption ($\boldsymbol{\lambda_{Aa}}$, $\boldsymbol{\lambda_{Ba}}$) and fission ($\boldsymbol{\lambda_{Af}}$, $\boldsymbol{\lambda_{Bf}}$) are coupled by two passage intensities

($\lambda_{At}$, $\lambda_{Bt}$) in two different directions. The model is shown in Fig. 2.

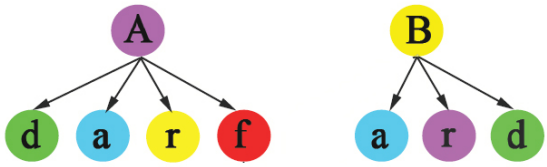


**Figure 2:** a two-point (one-group) model of various processes which particles can undergo.

In the model we include two sources of neutrons in two different regions with strength $S_A$ and $S_B$. The final expression of two-point (one-group) Feynman-alpha formulas is given below with source $S_B$:

$$\frac{\sigma_{ZZ}^2(t)}{\langle Z \rangle} = 1 + Y_1\left(1 - \frac{1-e^{-\omega_1 T}}{\omega_1 t}\right) + Y_2\left(1 - \frac{1-e^{-\omega_2 t}}{\omega_2 t}\right)$$

where

$$-\omega_1 = \frac{1}{2}\left(\lambda_{Af}\nu_A - \lambda_A - \lambda_B - \sqrt{(\lambda_{Af}\nu_A - \lambda_A + \lambda_B)^2 + 4\lambda_{At}\lambda_{Bt}}\right)$$
$$-\omega_2 = \frac{1}{2}\left(\lambda_{Af}\nu_A - \lambda_A - \lambda_B + \sqrt{(\lambda_{Af}\nu_A - \lambda_A + \lambda_B)^2 + 4\lambda_{At}\lambda_{Bt}}\right)$$

If detection is performed in the region A, than $Y_1$ and $Y_2$ functions should be used in the form like shown below:

$$Y_1 = \frac{2\lambda_{Ad}\left(\mu_{XX}\lambda_B + \mu_{XY}\lambda_{Bt} - \mu_{XX}\omega_1\right)\omega_1\omega_2}{S_B r' \lambda_{Bt}\omega_1\left(\omega_2 - \omega_1\right)}$$
$$Y_2 = \frac{2\lambda_{Ad}\left(\mu_{XX}\lambda_B + \mu_{XY}\lambda_{Bt} - \mu_{XX}\omega_2\right)\omega_1\omega_2}{S_B r' \lambda_{Bt}\omega_2\left(\omega_1 - \omega_2\right)}$$

However, if detector is placed in the region B, than following $Y_1$ and $Y_2$ functions must be used:

$$Y_1 = \frac{2\lambda_{Bd}\left[\mu_{YY}\lambda_A + \mu_{XY}\lambda_{At} - \mu_{YY}\left(\omega_1 + \lambda_{Af}\nu_A\right)\right]\omega_1\omega_2}{S_B r' \omega_1\left(\omega_2 - \omega_1\right)\left(\lambda_A - \lambda_{Af}\nu_A\right)}$$
$$Y_2 = \frac{2\lambda_{Bd}\left[\mu_{YY}\lambda_A + \mu_{XY}\lambda_{At} - \mu_{YY}\left(\omega_2 + \lambda_{Af}\nu_A\right)\right]\omega_1\omega_2}{S_B r' \omega_2\left(\omega_1 - \omega_2\right)\left(\lambda_A - \lambda_{Af}\nu_A\right)}$$

where

$$\mu_{XX} = -\frac{\lambda_{Af}\nu''_A\left(\lambda_B{}^2 + \omega_1\omega_2\right)\bar{N}_A + S_B\lambda_{Bt}^2 r''}{2\left(-\lambda_{Af}\nu_A + \lambda_A + \lambda_B\right)\omega_1\omega_2}$$
$$\mu_{XY} = -\frac{-\lambda_B\lambda_{Af}\lambda_{At}\nu''_A\bar{N}_A + S_B r''\lambda_{Af}\lambda_{Bt}\nu_A - \lambda_A S_B\lambda_{Bt} r''}{2\left(-\lambda_{Af}\nu_A + \lambda_A + \lambda_B\right)\omega_1\omega_2}$$
$$\mu_{YY} = \frac{S_B r''\left[\omega_1\omega_2 + \left(-\lambda_A + \lambda_{Af}\nu_A\right)^2\right] + \lambda_{Af}\lambda_{At}^2\nu''_A\bar{N}_A}{2\left(-\lambda_{Af}\nu_A + \lambda_A + \lambda_B\right)\omega_1\omega_2}$$
$$\bar{N}_A = \frac{S_B r'\lambda_{Bt}}{\omega_1\omega_2}$$

## 3. Simulation setup

In order to assess quantitatively the variance-to-mean formulas and their sensitivity to the isotopic content of nuclear fuel or waste, we performed a number of Monte-Carlo simulations [4].

The geometry of simulations is displayed in Fig. 3. The model used in the simulations consists of nuclear material (radius 4.46 cm), moderating material (thickness 21 cm) and two point detectors, placed in the regions A and B, respectively. The source with energy 2.5 MeV was placed at a distance of 15 cm from the centre of the nuclear material. Regions A and B in Fig.3 correspond to the nuclear material and moderator, respectively.

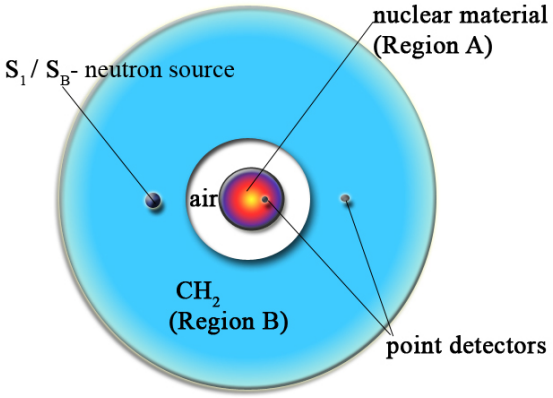


**Figure 3:** geometry used for the Monte-Carlo simulations.

In the simulations a few different compositions of the nuclear material were used. Each composition was denoted by a case number, as shown in Table 1. Apart of these cases one extra case (#6) was simulated when the composition of nuclear material were equivalent to the composition used in the case #5 but the moderating material in region B was removed.

| **Isotope** | $^{235}$U | $^{238}$U | $^{239}$Pu | $^{241}$Pu |
|---|---|---|---|---|
| **Case number** | **Amount of the isotope in nuclear material, %** | | | |
| **1** | 2.5 | 96 | 1 | 0.5 |
| **2** | 4 | 96 | 0 | 0 |
| **3** | 3.5 | 96 | 0.5 | 0 |
| **4** | 25 | 70 | 5 | 0 |
| **5** | 2.5 | 96 | 1.5 | 0 |

**Table 1:** different compositions of the nuclear material used in Monte-Carlo simulations.

Two different sets of data were simulated:

- reaction rates and reaction intensities obtained by review of the neutron weight balance table (print table 130);
- the flux of neutrons in point detectors (with 10 ns time interval).

The reaction intensities obtained from the simulations were normalized to one starting neutron and obtained with a relative error less than 1%.

From the first set of data transition probabilities/reaction intensities for two-point (one group) (see Table 2) and two-group (one-point) (see Table 3) cases were calculated.

The second set of data allowed to asses the Feynman-alpha curves in the way as it would be obtained in Feynman-alpha measurements.

| | Case number | | | | | |
|---|---|---|---|---|---|---|
| | **1** | **2** | **3** | **4** | **5** | **6** |
| **Region A** | | | | | | |
| $\lambda_f \cdot 10^2$ | 4.43 | 4.493 | 4.44 | 6.464 | 4.34 | 0.26 |
| $\lambda_a \cdot 10^2$ | 1.93 | 1.62 | 1.76 | 1.75 | 1.96 | 0.046 |
| $\lambda_d \cdot 10^4$ | 3.269 | 3.35 | 3.45 | 3.709 | 3.316 | 0.71 |
| $\lambda_t \cdot 10^1$ | 1.46 | 1.41 | 1.42 | 1.783 | 1.44 | 0.25 |
| $N$ | 2.68 | 2.47 | 2.545 | 2.59 | 2.66 | 2.7 |
| **Region B** | | | | | | |
| $\lambda_f$ | 0 | 0 | 0 | 0 | 0 | 0 |
| $\lambda_a \cdot 10^1$ | 9.73 | 9.67 | 9.68 | 1.0 | 9.7 | 0 |
| $\lambda_d \cdot 10^5$ | 8.41 | 7.9 | 8.27 | 9.86 | 8.368 | 0.43 |
| $\lambda_t \cdot 10^2$ | 9.11 | 9.119 | 9.09 | 9.343 | 9.09 | 0 |
| $v$ | 0 | 0 | 0 | 0 | 0 | 0 |

**Table 2:** transition probabilities / reaction intensities for two-region cases.

| | Case number | | | | | |
|---|---|---|---|---|---|---|
| | **1** | **2** | **3** | **4** | **5** | **6** |
| **Fast group (1)** | | | | | | |
| $\lambda_f \cdot 10^3$ | 8.28 | 7.68 | 7.84 | 21.5 | 7.97 | 2.686 |
| $\lambda_a \cdot 10^2$ | 1.32 | 1.37 | 1.41 | 1.53 | 1.39 | 0.0459 |
| $\lambda_d \cdot 10^4$ | 3.41 | 3.45 | 3.59 | 3.92 | 3.47 | 0.754 |
| $\lambda_t \cdot 10^1$ | 6.66 | 6.66 | 6.66 | 6.65 | 6.66 | 0 |
| $N$ | 2.68 | 2.47 | 2.545 | 2.59 | 2.66 | 2.7 |
| **Thermal group (2)** | | | | | | |
| $\lambda_f \cdot 10^2$ | 3.6 | 3.72 | 3.65 | 4.31 | 3.55 | 0 |
| $\lambda_a \cdot 10^1$ | 9.79 | 9.70 | 9.72 | 10.0 | 9.76 | 0 |
| $\lambda_d \cdot 10^5$ | 6.92 | 6.98 | 6.92 | 7.75 | 6.86 | 0 |
| $v$ | 2.68 | 2.47 | 2.545 | 2.59 | 2.66 | 2.7 |

**Table 3:** transition probabilities / reaction intensities for two-group cases.

## 4. Results and discussion

### 4.1. Theoretical evaluation

As a first step, evaluation of the variance to mean ratio was performed by using the values of the reaction intensities from Tables 2-3 together with formulas derived in Section 2 for two-point (one-group) and two-group (one-point) models.

As a result, $Y$-functions and $\omega$-time constants were calculated. The values obtained for the two-group (one-point) model are given in Tables 4-5, analogous values but for the two-point (one-group) model are shown in Tables 6-7.

| **Case number** | **Fast group (1)** | | **Thermal group (2)** | |
|---|---|---|---|---|
| | $Y_1 \cdot 10^5$ | $Y_2 \cdot 10^4$ | $Y_3 \cdot 10^6$ | $Y_4 \cdot 10^5$ |
| 1 | 1.035 | 1.248 | -4.929 | 2.29196 |
| 2 | 0.8698 | 1.006 | -4.218 | 1.88455 |
| 3 | 0.9677 | 1.126 | -4.451 | 2.00195 |
| 4 | 1.533 | 2.235 | -6.8 | 3.69 |
| 5 | 1.023 | 1.210 | -4.8 | 2.19 |

**Table 4:** the values of the $Y$-functions calculated by using the formulas from Subsection 2.1 for two-group (one-point) case.

| **Case number** | $\omega_1$ | $\omega_2$ |
|---|---|---|
| 1 | 1.14822 | 0.532483 |
| 2 | 1.13775 | 0.538275 |
| 3 | 1.1396 | 0.537353 |
| 4 | 1.18184 | 0.507841 |
| 5 | 1.14354 | 0.535041 |

**Table 5:** the values of the $\omega$-time constants calculated by using the formulas from Subsection 2.1 for two-group (one-point) case.

| **Case number** | **Region A** | | **Region B** | |
|---|---|---|---|---|
| | $Y_1 \cdot 10^6$ | $Y_2 \cdot 10^1$ | $Y_3 \cdot 10^7$ | $Y_4 \cdot 10^5$ |
| 1 | 1.41 | 0.10365 | -2.7 | 5.06 |
| 2 | 1.17 | 0.08644 | -1.9 | 3.62 |
| 3 | 1.28 | 0.09515 | -2.2 | 4.09 |
| 4 | 2.36 | 0.16104 | -5.5 | 11.4 |
| 5 | 1.36 | 0.1016 | -2.5 | 4.70 |

**Table 6:** the values of the $Y$-functions calculated by using the formulas from Subsection 2.2 for two-point (one-group) case.

| **Case number** | $\omega_1$ | $\omega_2$ |
|---|---|---|
| 1 | 1.07773 | 0.078298 |
| 2 | 1.07235 | 0.078592 |
| 3 | 1.07315 | 0.078563 |
| 4 | 1.11023 | 0.077139 |
| 5 | 1.07462 | 0.078169 |

**Table 7:** the values of the $\omega$-time constants calculated by using the formulas from Subsection 2.2 for two-point (one-group) case.

These coefficients allow to construct the dependence of the ratio of variance-to-mean of the number of detections on the detection time for two models and different compositions of nuclear material.

As an example, in Figure 4 one can see the Feynman-alpha curves of fast detections for five different compositions of nuclear material (see Table 1) if two-group (one-point) model is applied.

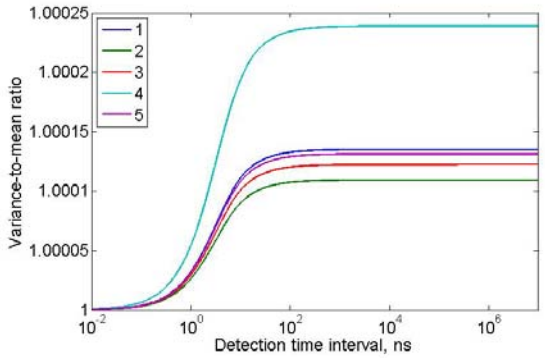


**Figure 4:** the dependence of the ratio of variance to mean of the number of **fast** detections on the detection time for five different compositions of nuclear material (**two-group (one-point) model**).

The presence of two exponents is not clearly visible in all cases. However, significant difference is observed between the theoretical curves for various concentrations of isotopes in nuclear material.

The biggest difference is observed for the case #4 (curve #4), which corresponds to the high enriched (~25 % $^{235}$U and ~5 % $^{239}$Pu) nuclear material.

Curve #1 lies under curve #4 but is above, even if very close to curve #5. The main difference between case #1 and #5 is the amount of $^{239}$Pu (~1 % $^{239}$Pu in the case #1 against ~1.5 % $^{239}$Pu in case #2) and the presence of $^{241}$Pu (~0.5 %) in the case #1.

Curve #3 lies below curve #5 and above curve #2. The reason for this is that in the case of #3, the amount of $^{239}$Pu is 3 times less compared to the case #5, whereas the content of $^{235}$U is 1.4 times higher in the case #3.

Similar results can be seen in Fig. 5 where Feynman-alpha curves of thermal detection in two-group (one-point) model are shown.

Thus, we can conclude that the two-group (one-point) Feynman-alpha model might be used in order to separate between the $^{239}$Pu, $^{241}$Pu and $^{235}$U contents.

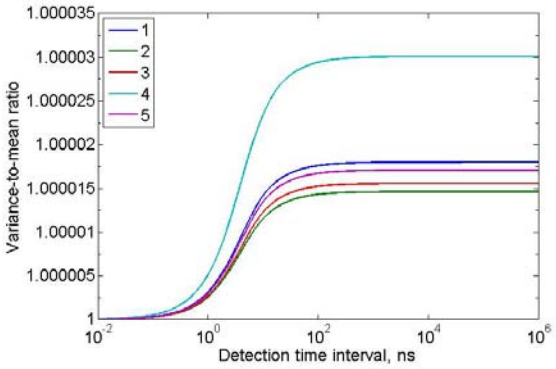


**Figure 5:** the dependence of the ratio of variance to mean of the number of **thermal** detections on the detection time for five different compositions of nuclear material (**two-group (one-point) model**).

It is interesting to notice that the two-point (one-group) model also allows separation between these isotopes, as can be seen in Fig. 6 (detection in region A) and Fig.7 (detection in region B).

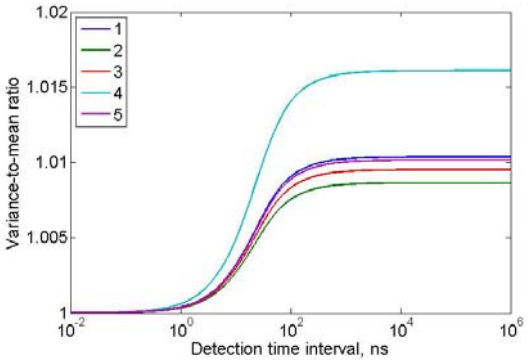


**Figure 6:** the dependence of the ratio of variance to mean of the number of detections **in region A** on the detection time for five different compositions of nuclear material (**two-point (one-group) model**).

However, the separation between the curves (different cases) is somewhat better when the Feynman-alpha curve for the detections in Region B is used, as one can see in Fig. 7 below, although the magnitude of the Feynman-alpha curves is lower in Region B compared to that in Region A.

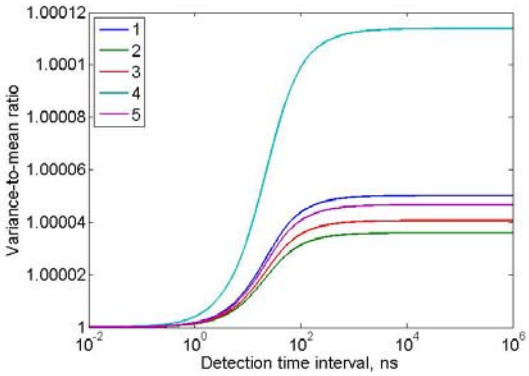


**Figure 7:** the dependence of the ratio of variance to mean of the number of detections **in region B** on the detection time for five different compositions of nuclear material (**two-point (one-group) model**).

## 4.2. Numerical evaluation

The situation will be different when we use the direct numerical evaluation of the Feynman-alpha curves.

First of all, as we can see in Fig. 8 if there is a moderator, i.e. presence of thermal neutrons in the system, then we can observe two plateaus in the curves. The first plateau is started in the time interval between 1-20 ns and it should be connected to the fast fissions happened in nuclear material. Second plateau is only appeared after $10^5$ ns (100 μs) that corresponds to the thermal fissions.

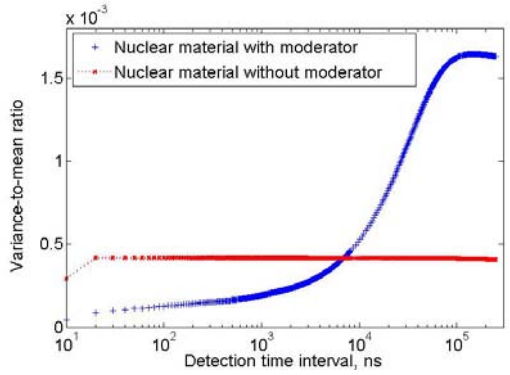


**Figure 8:** numerical Feynman-alpha curves obtained with detector placed in region A, composition of nuclear material corresponds to the case #5 and case #6.

Similar conclusions might be drawn if we consider the situation when fast and thermal neutron detectors placed in both regions at the same time for the same case, as is shown in Fig. 9.

In Fig. 9 we can see that curve (a), which corresponds to the Feynman-alpha curve for detections **in region A** in case of the **two-point (one-group) model,** is lying above curve (b), which relates to the same characteristics but for detections **in region B.** This observation agrees with theoretical results. However, as one can see, these two curves have two plateaus which are not visible in the theoretical plots.

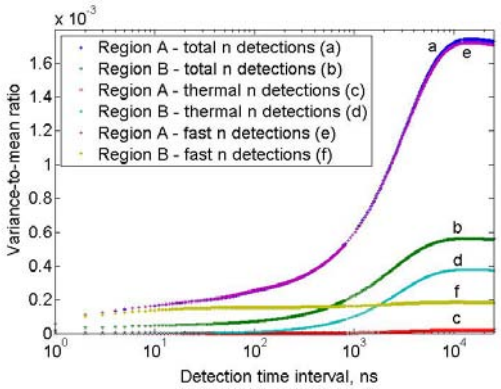


**Figure 9:** numerical Feynman-alpha curves obtained for a composition of nuclear material which corresponds to the case #4.

Now, let us consider the Feynman-alpha curves for detections in region A and B ("two-point (one-group) approximation") in connection to the various compositions of the nuclear material. As one can see in Fig. 10, there is an obvious difference between the curves for the case of the highly enriched nuclear material (case #4) and low enriched nuclear material, when the detector placed in the region A.

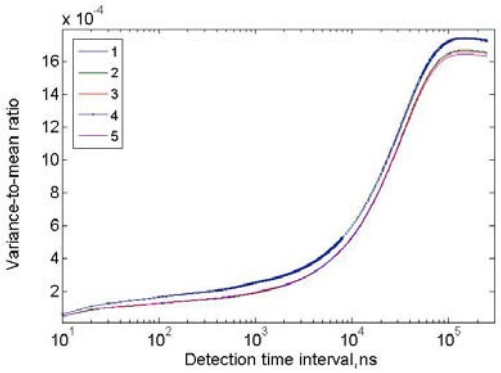


**Figure 10:** numerical Feynman-alpha curves obtained with detector placed in region A, for five different compositions of nuclear material.

However, when the detector is placed in region B (Fig. 11) the slight difference between these cases is seen only for the second plateau, while in the previous Fig. 10 both plateaus were affected by the change of the composition of nuclear material.

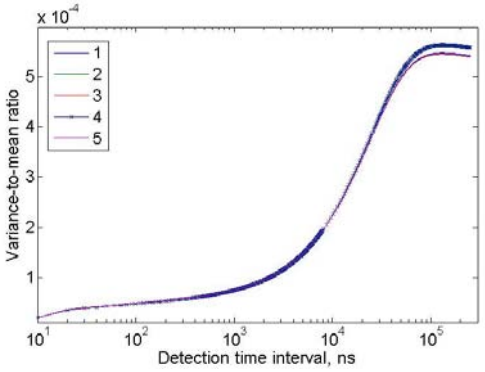


**Figure 11:** numerical Feynman-alpha curves obtained with detector placed in region B, for five different compositions of nuclear material.

Thus, we can conclude that the two-point approximation might be sufficiently good to differentiate between highly enriched and low enriched nuclear material. However, it is not very sensitive to the slight changes in the isotopic composition of nuclear material. Though, it follows the same trend as theoretical curves, i.e. top curve corresponds to the case # 4, after that cases #1, #5, #3 and #2 follow sequentially.

The opposite situation is observed when we use detection of the thermal neutrons in region A (similar to the two-point two-group approximation). The Feynman-alpha curves are very different for various composition of nuclear material, as shown in Fig. 12.

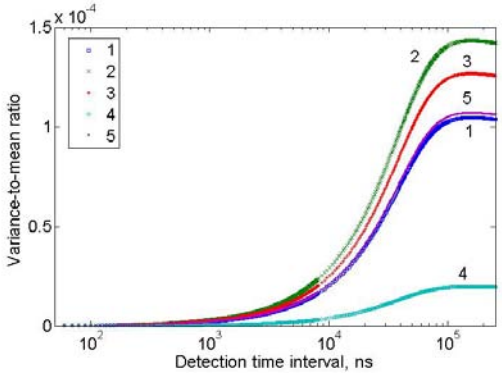


**Figure 12:** numerical Feynman-alpha curves obtained with detector of thermal neutrons, placed in region A, for five different compositions of nuclear material.

It is interesting to note that in Fig. 12 one can observe a reverse behavior of numerical Feynman-alpha curves compared to what we have seen in Fig. 5 (theoretical curves) for five different compositions of nuclear material. The top curve corresponds to case #2 (~4 % $^{235}$U and ~96 % $^{238}$U) while the bottom curve #4 is related to the highly enriched material (~25 % $^{235}$U and ~5 % $^{239}$Pu). At the same time we should mention that all curves in Fig.12 have only one plateau which appears after $5 \cdot 10^4$ ns (50 μs).

### 4.3. Numerical evaluation (ADS)

The main focus of this paper was directed to the evaluation and use of two-point (one-group) and two-group (one-point) Feynman-alpha theories in the Safeguards applications. However, it is interesting also to have a look to the Feynman-alpha curves which can be obtained in ADS.

Therefore, we run the simulations for the real prototype of ADS [5] in the same way, as described in Section 3, but using the real geometry from report [5]. Two point detectors were place in the core and reflector. The results of simulation are shown in Fig. 13.

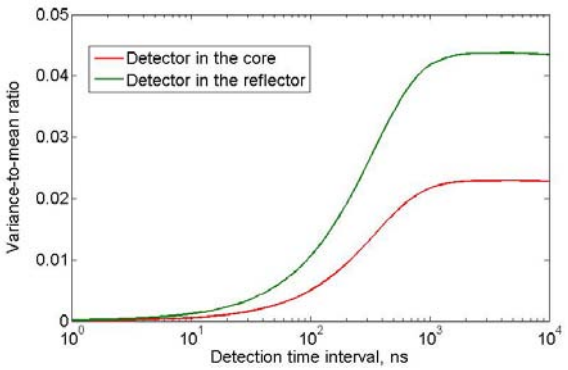


**Figure 13:** numerical Feynman-alpha curves obtained with detectors placed in the core and reflector of ADS [5].

As one can see in Fig. 13, in both regions the only one plateau is observed, which can be explained by the prevalent presence of fast neutrons.

## 5. Conclusions

In the present study we have extended the two-group (one-point) and two-point (one-group) versions of Feynman-alpha theory. The sensitivity of Feynman-alpha approach to various isotopic content of nuclear fuel or waste was also evaluated in two ways, theoretically and numerically.

It was observed that presence of two exponents in the dependence of the ratio of variance-to-mean of the number of detections on the detection time is not clearly visible for all theoretical cases in Safeguards application. The same situation is seen for numerical

Feynman-alpha curves obtained with detectors placed in the core and reflector of ADS.

However, there is a big difference between the theoretical curves observed for various concentrations of isotopes in nuclear material. The biggest difference is observed for the case which corresponds to the high enriched (~25 % $^{235}$U and ~5 % $^{239}$Pu) nuclear material.

Results of the calculations indicate that the two-group (one-point) and two-point (one-group) Feynman-alpha theoretical model might be used in order to separate between the $^{239}$Pu, $^{241}$Pu and $^{235}$U.

However, in two-point (one-group) Feynman-alpha theoretical model, the separation between the curves (different compositions of nuclear material) is somewhat better when the Feynman-alpha curve for the detections in Region B is used. Though, the magnitude of Feynman-alpha curves is lower in Region B compared to the Region A.

On the other hand, the results of the numerical evaluation of the Feynman-alpha curves do not fully agree with theoretical models. In the case when there is a moderator in the system, i.e. presence of thermal neutrons, it is possible to separate between two plateaus in the curves. The first plateau is started in the time interval between 1-20 ns and it is connected to the fast fissions happened in nuclear material. The second plateau only appears after $10^5$ ns (100 μs) that corresponds to the thermal fissions.

In connection to the various isotopic compositions of nuclear fuel, two-point approximation might be sufficiently good to differentiate between highly enriched and low enriched nuclear material. However, it is not very sensitive to the slight changes in the isotopic composition, even if it follows the same trend as the theoretical curves.

The opposite situation is observed when a detection of thermal neutrons in region A is used. Then, the Feynman-alpha curves are very different for various compositions of nuclear material, although the behavior of these numerical Feynman-alpha curves is somewhat reversed for different compositions compared to the theoretical predictions.

It is worth to mention that additional studies are needed in order to further evaluate the sensitivity of the Feynman-alpha approach to the isotopic content of nuclear fuel, which can be useful for Safeguards and online evaluation of the fuel content in ADS systems, for example.

## Acknowledgement

This work was supported by the **Swedish Radiation Safety Authority, SSM**.

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# Burnup monitoring of spent fuel assemblies

**I. Almási, C.T. Nguyen, Z. Hlavathy, J. Zsigrai[1] L. Lakosi, P. Nagy**

Centre for Energy Research
Hungarian Academy of Sciences
29-33 Konkoly-Thege M. u., Budapest, 1121 Hungary
E-mail: istvan.almasi@energia.mta.hu, laszlo.lakosi@ energia.mta.hu

***Abstract:***

*Spent VVER-440 fuel assemblies submerged in the loading pit were measured by high-resolution gamma spectrometry. The assemblies were moved to the front of a 1.2 m long collimator tube built in the concrete wall of the pit in the reactor block at the Paks Nuclear Power Plant, and lifted down and up under water for scanning by the refuelling machine. The HPGe detector was placed behind the collimator in an outside staircase. The measurements involved scanning of the assemblies along their length of all the 6 sides, at 5-12 measurement positions side by side. Axial and azimuthal burnup profiles were taken in this way. Assembly groups for measurements were selected according to their burnup (10 – 45 GWd/tU) and special positions (e. g. control assembly, neighbour of control assembly). Burnup differences were well observable between assembly sides looking at the center of the core and opposite directions. Also, burnup profiles were different for control assemblies and normal fuel assemblies. The ratio of the measured activities of Cs-134 and Cs-137 was evaluated by relative efficiency (intrinsic) calibration. Measurement uncertainty is around 3%. Taking into account irradiation history and cooling time, the activity ratio Cs-134/Cs-137 shows good correlation with the declared burnup.*

**Keywords:** Cs-134/Cs-137 ratio; high resolution gamma spectrometry; non-destructive assay; spent fuel verification

## 1. Introduction

The objective of this project is to support burnup calculation of spent VVER-440 type fuel assemblies at Paks Nuclear Power Plant (NPP) by an independent experimental method. Burnup is an important safeguards issue. In addition, burnup data are useful for the assessment of fuel performance in the core as well as for reactor safety and fuel economy considerations. Exploitation of the nuclear fuel more effectively is limited by the uncertainty of the burnup calculation code. If the uncertainty of calculation were less, the safety margin could be decreased, thus fuel burnup could be raised for more effective fuel use (in addition to introduction of new assembly types and longer refuelling period). The burnup limit is provided by the supplier of the fuel. This value is to be reduced by the uncertainty of the burnup determination. The limit thus obtained should be observed during reactor operation. Producing more power means that fuel burnup should increase, otherwise the amount of fuel should be raised thereby worsening the economy.
To validate the burnup calculation from reactor parameters, high resolution gamma spectrometric (HRGS) measurements have started for assessing the uncertainty of the burnup code used at the NPP. This paper contains a comparison of the measured and simulated values of the activity ratio of the fission products Cs-134 and Cs-137. Exact correction of the calculated burnup based on the data presented here will be done in the future.

## 2. Experimental

[1]Present address: EC, JRC, Institute of Transuranium Elements, Karlsruhe, Germany

The fuel assemblies were transferred under water to the loading pit for measurement. The experimental setup, shown in Fig. 1, was the same as described in an earlier paper [1]. The assemblies were moved to the front of a lead collimator built into the concrete wall of the loading pit in the reactor block, and moved down and up under water by the refuelling machine. The 120 cm long collimator tube was shut on the water side by a 5 cm thick steel plate. Translation and rotation of the assemblies were performed for a full scanning. The distance of the assemblies from the closing plate of the collimator was about 15 cm in water. Gamma spectra were taken by a 46 $cm^3$ coaxial HPGe detector placed behind the collimator in an outside staircase. The collimator enabled the detector to see the whole cross section of the assemblies, in 1 cm height. Remote control by a laptop computer was done from the measurement stand in the reactor hall to another laptop PC controlling the multichannel analyzer in the staircase through a local area network. Measurements were performed at 5-12 measurement positions along each of the 6 sides of the hexahedral prism shape assemblies.

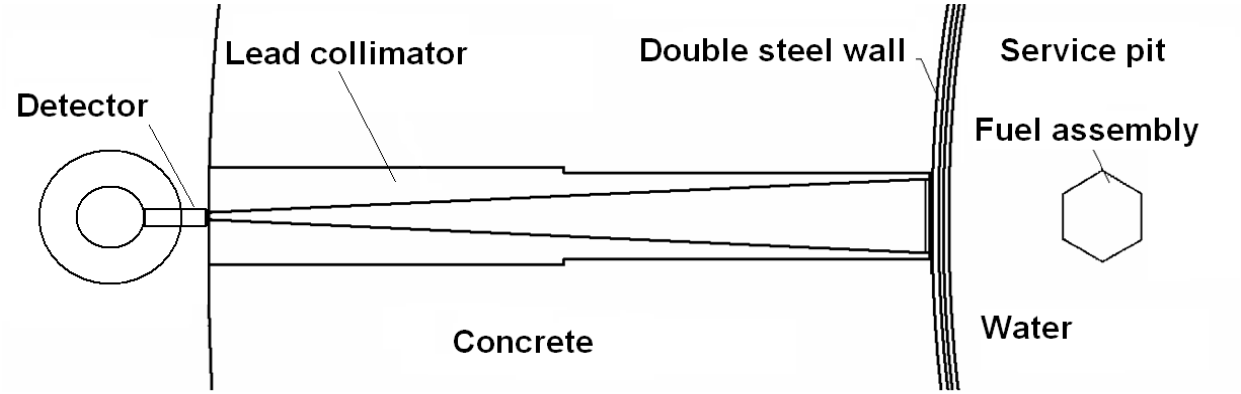


**Fig. 1**. Scheme of the experimental setup

It is well known that the activity of the fission product Cs-137 is proportional to the burnup (see, e. g., [2]). The measured Cs-137 intensity depends on the geometry as well. The correlation between burnup and the count rate of $^{137}Cs$ measured at half-height of the assemblies is shown in Fig. 2.

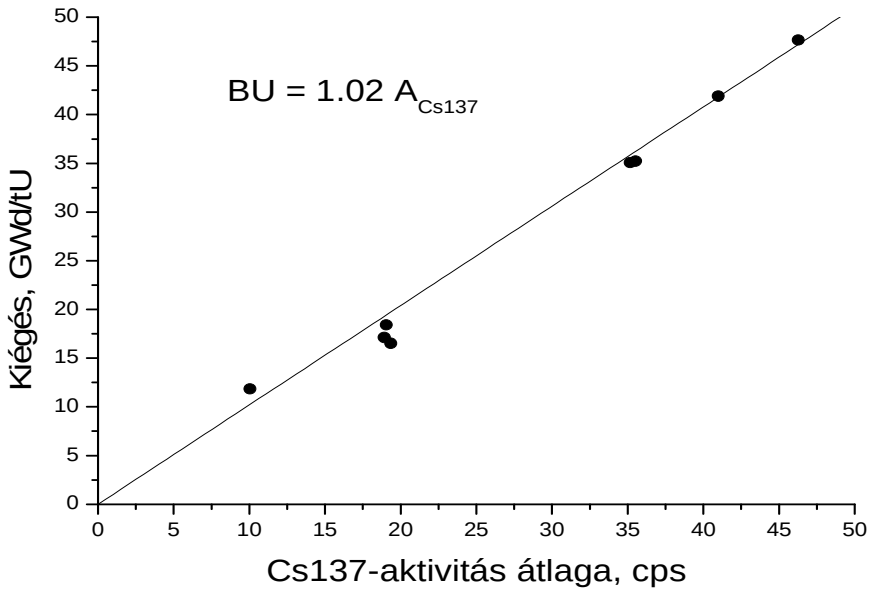


**Fig.2.** Correlation between burnup and the count rate of $^{137}Cs$ measured at half-height of the assemblies

A typical axial distribution of the count rate of the 662 keV gamma rays of Cs-137 and the calculated burnup profile are plotted in Fig. 3.

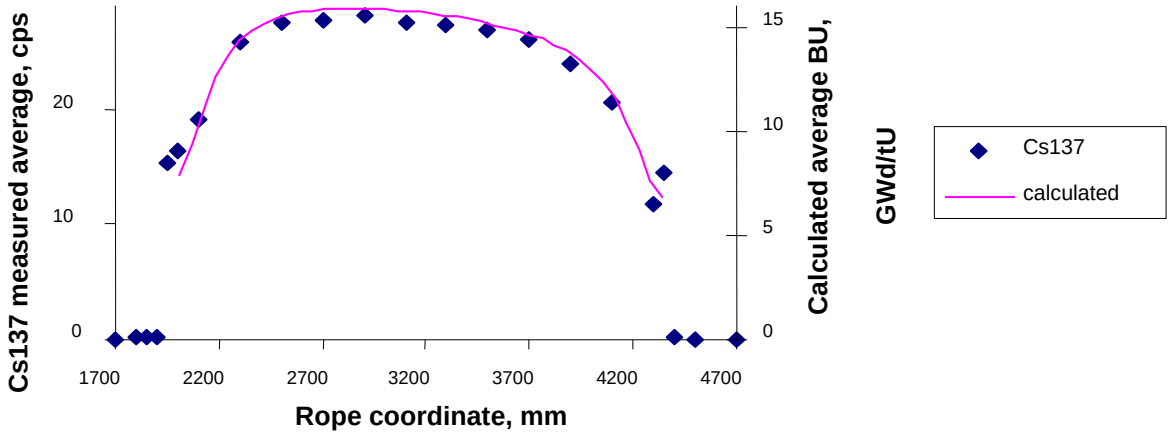


**Fig. 3.** A typical measured axial Cs-137 activity profile and the calculated burnup profile

Whereas the intensity profile along the length of an assembly follows the *axial* burnup profile, intensity values, however, depend on the measurement geometry, the absorption of gamma-rays, and the detector applied. Furthermore, incorrect assembly positioning is much more likely during assembly rotation than during axial scanning. Since experimental conditions did not allow arranging the geometry precisely enough, Cs-137 activity alone is not suitable for examining the *azimuthal* burnup profile, for which the activity ratio Cs-134/Cs-137 is more suitable. As Cs-134 is not a direct fission product, but is formed by neutron capture on the fission product Cs-133, its abundance is approximately proportional to square of the burn-up. The activity ratio of Cs-134 to Cs-137 (Cs ratio) is therefore proportional to the burnup again [3], and the geometry factor is thus eliminated. For this reason the ratio can be used for estimating burnup. Unlike the count rate of a given peak, the activity ratio is independent of the geometry, the absorption, and the detector, therefore it finely follows the azimuthal profile of the assembly burnup. For illustration of the independence of the Cs ratio on the measurement geometry, an assembly was measured at different distances from the collimator. The dependence of the Cs ratio on the distance is shown in Fig. 4. As seen, the measured ratio remains unchanged within to 1 % at the assembly movement of more than 20 cm.

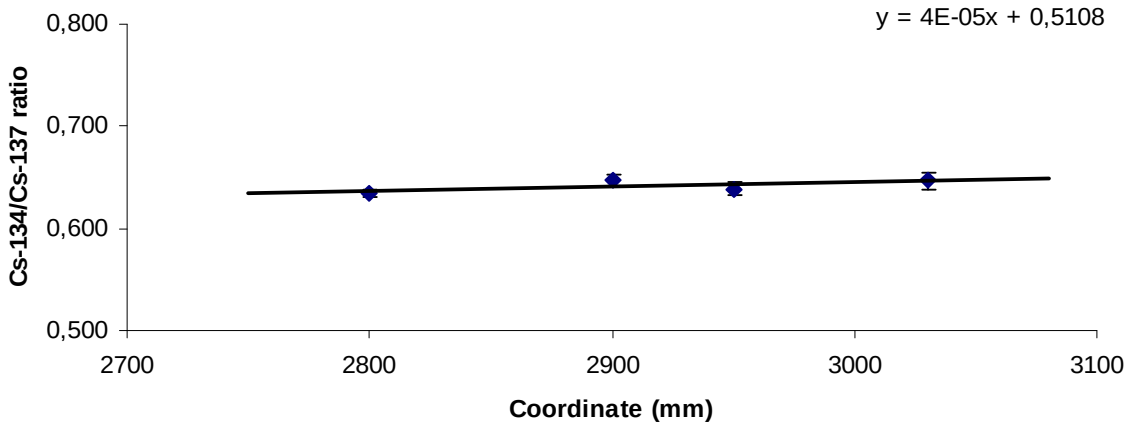


**Fig. 4.** Activity ratio Cs-134 / Cs-137 as a function of the assembly-to-detector horizontal distance

A total of 28 spent fuel assemblies were measured at 5-12 fixed height positions, from all six sides. Four assembly groups were selected, six assemblies in each, as the core was divided into six sectors. From all sectors, assemblies of similar positions and histories were chosen. Their burnup and cooling time varied between 10 – 44 GWd/tU and between 0.5 – 5.5 years, respectively. Follower parts of two control assemblies and their neighbours were also assayed. The intensities of the 605 and 796 keV peaks of Cs-134 as well as the 662 keV peak of Cs-137 were measured for 300 – 600 s in each position.
The measurement campaign lasted for 5 weeks. About 1200 spectra were evaluated by dedicated software developed at the institute. Statistical uncertainty of peak areas was around 1%. Cs ratios were extrapolated to a common date of the beginning of the measurements, due to the relatively short half-life (2.06 y) of Cs-134, and were determined for each measurement point. The extrapolated Cs activity ratios varied between 0.1 and 1.5.

## 3. Evaluation and results

Activity ratios were determined using the main gamma lines of Cs-134 (605, 795 keV) and the 662 keV line of Cs-137 by the relative efficiency (intrinsic) calibration method [3]-[5], which utilizes the fact that activities calculated from different gamma lines of the same isotope must be equal. Cs-137 activity and the activity ratio were evaluated at every measurement point.
The relative efficiency curve was constructed by using the two gamma lines of Cs-134 mentioned above. Using intrinsic calibration, the relative efficiency $\varepsilon_{Cs134}(E)$ as a function of the gamma-ray energy $E$ can be given as

$$\varepsilon_{Cs134}(E) = \frac{(I_{796}/B_{796} - I_{605}/B_{605})}{796keV - 605keV} \times E(keV) + \frac{I_{605}/B_{605} \times 796keV - I_{796}/B_{796} \times 605keV}{796keV - 605keV}. \quad (1)$$

Then the activity ratio $^{134}Cs/^{137}Cs$ can be determined as

$$\frac{A_{134}}{A_{137}} = \frac{\varepsilon_{Cs134}(662keV)}{I_{662}/B_{662}} \ . \quad (2)$$

Here $I_{605}$ and $I_{796}$ are the measured peak areas of the 605 and 796 keV γ-lines of $^{134}Cs$, $I_{662}$ is that of the 662 keV γ-line of $^{137}Cs$, whereas $B_{605}$, $B_{796}$, and $B_{662}$ are the corresponding emission probabilities. For energies above ~300 keV, the efficiency curve of HPGe detectors can be approximated with a linear function on a log-log scale. In the short interval between 605 and 796 keV, the linear approximation gives satisfactory accuracy even on a linear scale, as above in Eq. (1). Using the γ-energies 605, 662, and 796 keV, a Monte Carlo simulation provided a difference below 2% from the linear approximation in the relevant energy interval. For real spectra the statistical uncertainty of the peaks other than those at 605 and 796 keV is quite high, thus only the peaks at 605 and 796 keV were used for constructing the relative efficiency curve. Using only two points on the linear scale simplifies the calculations, and provides an efficiency curve which might be the cause of small systematic biases in the measured Cs-ratio, but is free from the random uncertainties which would emerge from using more gamma energies and a more complicated functional form for the efficiency. The eventual systematic bias is expected to be well below the uncertainty which is aimed at the present stage of the project.
The measured and calculated burnup profile of an assembly in terms of the Cs ratio is shown in Fig. 5. Burnup asymmetries throughout the core were revealed and interpreted from the reactor physical point of view. Four groups of assemblies were measured with various burnup and cooling time. Differences of uniform burnup assemblies were observable between assembly sides looking at the center of the reactor core and opposite directions. This is illustrated on a polar diagram in Fig. 6 by a graphical representation of the Cs ratio measured in a group of six assemblies symmetrically positioned around the core during reactor operation. The burnup of the assemblies was uniformly 39 GWd/tU, whereas they were measured with 1.5 year cooling. It was shown that Cs ratios (in blue) are higher on the assembly sides looking at the core center (the values plotted at the vertices of the hexagons correspond to those at faces of the assemblies). In this case Cs ratio is varying between 0.88 – 0.92 and 1.0 – 1.03, reaching the higher value on the side nearer the core center and the lower value on the opposite side.

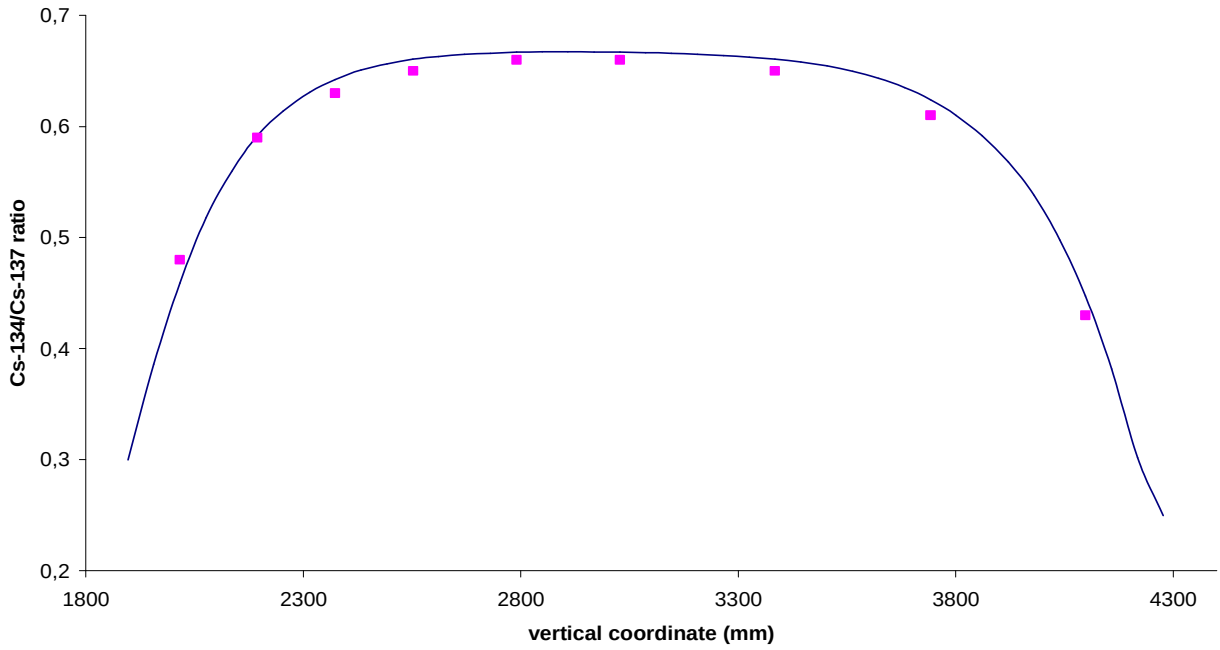


**Fig. 5.** A typical Cs ratio profile with experimental points and theoretical curve

For example, Cs ratios measured on the 6 sides of the assembly on the top of the figure are as follows: on the 3 sides looking straight down (direction core center) and sidewards–down, they are around 1.0, whereas on the 3 sides looking straight up and sidewards-up, around 0.9.

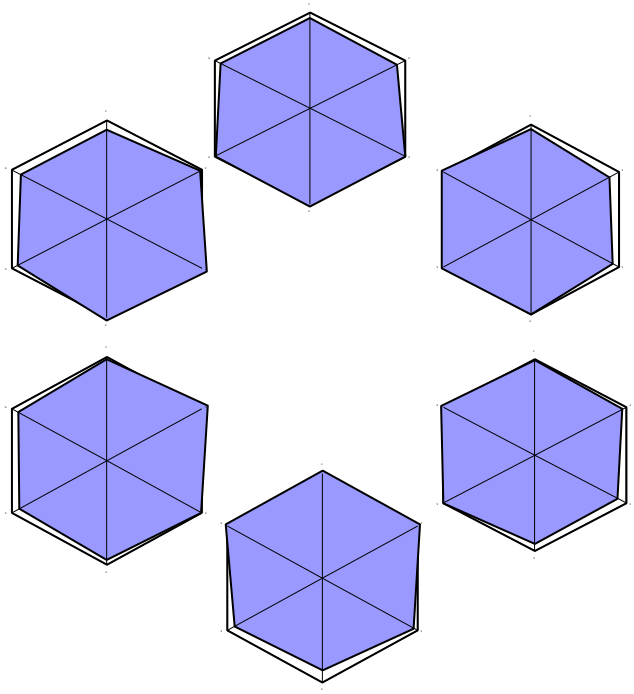

**Fig. 6.** Measured azimuthal Cs-ratio profiles at half-height of pins of the 39 GWd/tU burn-up assembly group around the core

Certain groups of assemblies show strong burnup asymmetries depending on the position of their sides with respect to the centre of the core, referring to “anomalies” in temperature or neutron flux in the core. Burnup profiles of the working assemblies and of the followers also differ from each other, in accordance with the measurements. Measured and calculated axial profile (in terms of Cs ratio) of a working assembly is plotted in Fig. 7, whereas that of a control assembly is plotted in Fig. 8, at various turning angles.

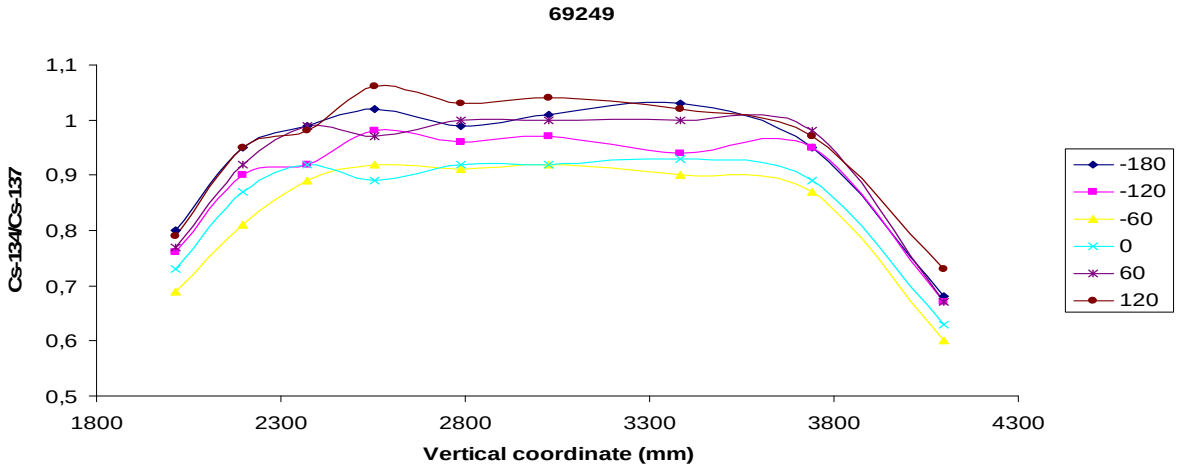


**Fig 7.** Calculated and measured axial burnup profiles of a working assembly at various turning angles

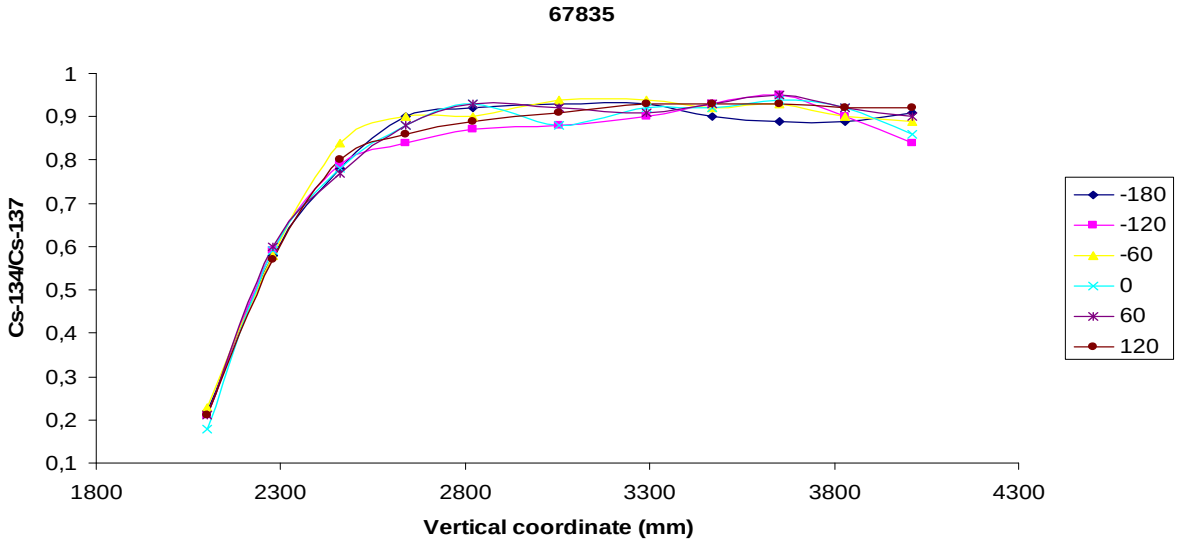


**Fig. 8.** Calculated and measured axial burnup profiles of a control assembly at various turning angles

Experiments to follow are required to refine the agreement between experiment and calculation, thus further decrease the uncertainty.

## 4. Conclusions

Both the axial and azimuthal burnup profile of spent fuel assemblies can be followed using the results from non-destructive HRGS measurements.
Uncertainty of the measured data is about 3 %. This can be improved by longer measurements. We would eventually like to reach 1 % accuracy at least.

In this initial state of the project it is expected that the uncertainty of burnup calculation based on the error of power calculation can be decreased by utilizing the data of the experimental results. This facilitates observing safety limits more accurately in planning of the loading pattern of the core and during reactor operation.
Measurement of additional assemblies is needed to refine the agreement between experiment and calculations.
It is foreseen that the experimental results will be used to improve the accuracy of the burnup calculation code, thus enabling more efficient use of the nuclear fuel. However, the connection between the uncertainty of the burnup value and the Cs activity ratio needs to be investigated for different operational parameters and cooling times.

## 5. Acknowledgements

The authors would like to acknowledge the sponsorship of Paks NPP and its personnel for assisting at the measurements as well as Messr. T. Parkó and I. Pós at the NPP for performing burnup and Cs ratio calculations of the measured assemblies. Also, the project was partly supported by the Hungarian Atomic Energy Authority. The authors are greatly indebted to Mr. J. Huszti for developing the dedicated software package.

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Contribution to the ESARDA Symposium in Bruges, Belgium, May 2013

# Planning Interim Inspections: The Role of Information

**Rudolf Avenhaus**
Universität der Bundeswehr München
Werner-Heisenberg-Weg 39
85579 Neubiberg, Germany
rudolf.avenhaus@unibw.de

**Thomas Krieger**[1]
Ramperstorffergasse 38 / 5+6
1050 Wien, Austria
krieger.thomas@gmx.net

***Abstract:***

*When interim inspections in nuclear installations are planned, it has to be decided, if the time points of all inspections are fixed at the beginning of the reference time interval, e.g., one year, or if they are fixed sequentially. In the latter case the time point for the second inspection is fixed only after the first one has been performed, the third after the second one, and so on. For that decision not only organisational aspects have to be taken into account, but also the role of information: Will the Inspectorate in the latter case be able to draw an advantage from the fact that after the first inspection it may know what the operator´s behaviour was so far? Vice versa, the same holds for the operator in case he plans to start an illegal activity in the course of the reference time interval.*

*In this contribution these questions will be answered with the help of simplified inspection schemes and game theoretical arguments. It is shown that it depends on the parameters of the model, e.g., detection probabilities, whether or not sequential procedures are meaningful in the sense that the optimal expected detection time is shortened, and it is tried to develop an intuitive understanding of these results which may provide help for more complicated situations.*

**Keywords:** EURATOM; IAEA; Inspection planning; Detection time; Error of the second kind

## 1. Problem Formulation

Short notice and unannounced interim inspections (in the following shortly interim inspections) are a useful safeguards tool because they may shorten the detection time, i.e., the time between the start of an illegal activity and its detection. More than that because of their unpredictability they may help to deter any person or organisation from illegal behaviour. Therefore the problem of interim inspections to be carried through both by EURATOM and the IAEA has been discussed for many years; it has been settled only some time ago in the so-called IAEA/EURATOM Partnership Approach [1].

[1] This work has been completed during this author´s affiliation with the Universität der Bundeswehr München, Neubiberg, Germany.

On the other hand unpredictability cannot mean arbitrariness, on the contrary, international safeguards have to be performed in line with agreed rules which mean that these inspections have to be planned and carried through accordingly: they have to be arranged on the base of probabilistic objectives, for example expected detection times.

Interim inspections pose specific problems different from standard safeguards measures like material accountancy and data verification. While the standard measures provide very detailed information at the end of a reference time interval, and while they are characterized by the use of advanced statistical techniques for the compilation and evaluation of measurement data, interim inspections are aiming at the immediate detection of illegal activities or, positively formulated, confirmation of legal behaviour. Therefore, primarily simple techniques for the checking of seals, or comparing installations in facilities with the design information provided by the facility attachments, are used, and above all, time is important: Time available for the inspector in the facility, and time elapsed between the start of an illegal activity and its detection.

In order to be able to perform substantive analyses of interim inspections many assumptions have to be made which may be disputed since they are either not explicitly formulated in available nuclear safeguards documents or deal with the behaviour of facility operators, organisations or States in case they might behave illegally. In a previous study [2] these assumptions have been classified and it was shown how many different quantitative models would have to be studied if one would try to analyse the consequences of all these assumptions.

In this contribution one aspect of the planning of interim inspections is studied in detail: Should the responsible Inspectorate fix the time points of all inspections in one nuclear facility and in a given reference time interval right at the beginning of this interval, or else, should it fix at the beginning only the time point of the first inspection, after that inspection that of the second one and so on. In doing so, the operator´s behaviour, should he try to act illegally, has to be taken into account as indicated above.

Since the analysis of these problems turns out to pose major mathematical difficulties for any number of interim inspections in a reference time interval, only the most simple case is considered here: Only two inspections will be taken into account, and more than that, these two inspections can be performed only at three distinct points of time. It will turn out that even under these very limiting assumptions the analysis is not trivial and very interesting results are obtained.

## 2. Game Theoretical Model and Analysis

Let us assume that in a nuclear facility *two interim inspections* will be performed by an international safeguards organisation (in the following shortly inspector) in a reference time interval at the beginning and end of which a physical inventory verification (PIV) takes place. There are only three time points 1, 2 and 3 available for these two interim inspections, see Figure 1.

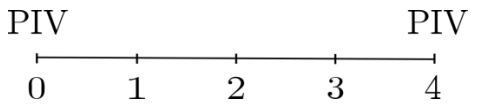


Figure 1: Possible time points for interim inspections.

The operator of the facility will start exactly one illegal activity within the reference time interval, this assumption has been discussed in [2] as well – at time points $0$,$1$ or $2$. The choice of time point $3$ will not be considered here, since it is weakly dominated by the other ones, see, e.g., [3]. The illegal activity will be detected with probability $1 - \beta$ by the next inspection or, if not earlier, with certainty at the end of the reference time interval, i.e., in case of coincidence of an inspection and the start of the illegal activity, it will be detected only at the occasion of the next inspection or the PIV. Also only attribute sampling inspection procedures are considered which means that the possibility of errors of the first kind, i.e., false alarms, is excluded. The payoff to the operator is the time elapsed between the start of the illegal activity and its detection, and the payoff to the inspector is its negative value, in other words, non-cooperative two person zero sum games are considered here.

Four variants of this general model are analysed: The operator may decide at the beginning, i.e., at time point 0, when to start his illegal activity, or he may only decide whether to start the illegal activity immediately or to postpone it; in the latter case he may decide again after the first inspection, and so on. The inspector may decide at time point 0 when to perform both inspections, or he may decide only when to perform the first inspection, and after the first one when to perform the second one. Calling the behaviour of both players sequential or non-sequential, there are four variants No-No, No-Se, Se-No and Se-Se as indicated in Table 1.

| Inspector / Operator | Non-sequential | Sequential |
|---|---|---|
| Non-sequential | No-No | No-Se |
| Sequential | Se-No | Se-Se |

Table 1: Four variants of the general inspection game and their abbreviations.

It should be mentioned that the general model is the simplest model with the help of which the impact of the above described behaviour – sequential or not – can be studied since, if there is only one inspection, there is no difference between the four variants. Nevertheless, there exists a realistic application of this model, see [2].

In the following all four model variants will be analysed in detail and their solutions – optimal strategies of both players and optimal expected detection times – will be presented in the form of four Theorems. In the third chapter these solutions will be compared; their meaning for practical applications will be discussed in the concluding chapter.

## 2.1. The No-No Variant

As mentioned above, in this variant the operator decides at time point $0$ when to start his illegal activity, namely at $0$ or at $1$ or at $2$. Also at time point $0$, and without knowledge of the operator's decision, the inspector decides when to perform his two inspections, namely at $(1,2)$ or at $(1,3)$ or at $(2,3)$. The game ends if the illegal activity is detected in the course of an interim inspection or at the latest at time point $4$, i.e., at the PIV.

The extensive form of this variant, which has been analysed already earlier, see [2], is represented graphically in Figure 2.

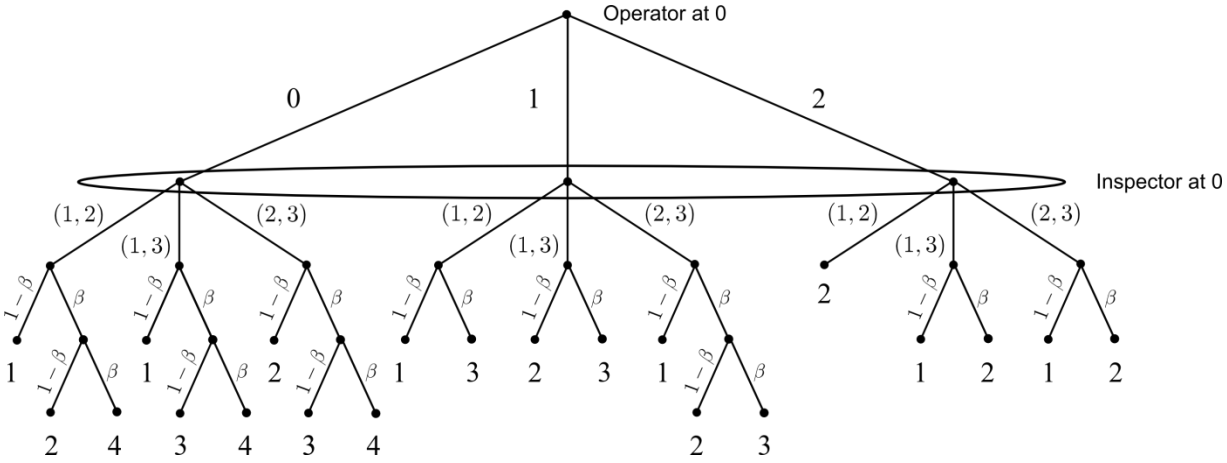


Figure 2: Graphical representation of the extensive form of the No-No variant of the inspection game. The encircled area denotes the information set of the inspector.

In order to model the information situation of the inspector at $0$, namely that he does not know the operator's decision at $0$, the concept of the information set is used: He cannot distinguish between nodes of the same information set, i.e., he does not know – according to our model assumptions – the start point of the illegal activity. At the end nodes of the game tree the detection times are drawn. It should be mentioned that this variant can also be represented in a natural way in normal form; the extensive form is preferred here since it can better be compared with those of the other variants.

Let $\boldsymbol{q} = (q_0, q_1, q_2)$ with $q_0 + q_1 + q_2 = 1$ be the operator´s probabilities to choose the time points $0, 1$ or $2$ for the start of the illegal activity, and let $\boldsymbol{p} = (p_{(1,2)}, p_{(1,3)}, p_{(2,3)})$ with $p_{(1,2)} + p_{(1,3)} + p_{(2,3)} = 1$ be the inspector´s probabilities to choose the pairs of time points $(1,2)$, $(1,3)$ and $(2,3)$ for his interim inspections. Then the expected payoff to the operator, i.e., the overall expected detection time is

$$\begin{aligned} E_{No-No}(\boldsymbol{q},\boldsymbol{p}) &= q_0\,[\,p_{(1,2)}\,(1+\beta+2\,\beta^2) + p_{(1,3)}\,(1+2\,\beta+\beta^2) + p_{(2,3)}\,(2+\beta+\beta^2)\,] \\ &+ q_1\,[\,p_{(1,2)}\,(1+2\,\beta) + p_{(1,3)}\,(2+\beta) + p_{(2,3)}\,(1+\beta+\beta^2)\,] \\ &+ q_2\,[\,p_{(1,2)}\,2 + p_{(1,3)}\,(1+\beta) + p_{(2,3)}\,(1+\beta)\,]\,. \end{aligned} \tag{1}$$

We are looking for the solution of this game which is given by the Nash equilibrium [4] with respect to $\boldsymbol{q}$ and $\boldsymbol{p}$ which says that any unilateral deviation from the equilibrium strategy does not improve the deviator's payoff. In the case of the zero sum game considered here and subsequently, the equilibrium strategies $\boldsymbol{q}^*$ and $\boldsymbol{p}^*$, also called *optimal strategies*, are determined with the help of the saddle point criterion

$$E_{No-No}(\boldsymbol{q},\boldsymbol{p}^*) \le E^* \le E_{No-No}(\boldsymbol{q}^*,\boldsymbol{p}) \quad \text{for all} \quad \boldsymbol{q} \ \text{and} \ \boldsymbol{p} \tag{2}$$

where $E^*_{No-No} = E_{No-No}(\boldsymbol{q}^*,\boldsymbol{p}^*)$ is the *optimal expected detection time*. We formulate the result as

**Theorem 1:** Given the non-cooperative two person zero sum game the extensive form of which is represented graphically in Figure 2. The solution of this game, i.e., the optimal strategies $\boldsymbol{q}^* = (q_0^*, q_1^*, q_2^*)$ and $\boldsymbol{p}^* = \left(p^*_{(1,2)}, p^*_{(1,3)}, p^*_{(2,3)}\right)$ and the optimal expected detection time $E^*_{No-No}$, is given as follows:

| | $0 \le \beta \le 1/2$ | $1/2 < \beta \le 1$ | |
|---|---|---|---|
| $\boldsymbol{q}^*$ | $\frac{1}{3+2\,\beta+\beta^2}\,(1+\beta\,,1\,,1+\beta+\beta^2)$ | $(1,0,0)$ | (3) |
| $\boldsymbol{p}^*$ | $\frac{1}{1-\beta}\left(\frac{1+\beta+2\,\beta^2+\beta^3}{3+2\,\beta+\beta^2}\,,\ \frac{(1-2\,\beta)(1+\beta+\beta^2)}{3+2\,\beta+\beta^2}\,,\ \frac{(1-2\,\beta)(1+\beta)}{3+2\,\beta+\beta^2}\right)$ | $(1,0,0)$ | (4) |
| $E^*_{No-No}$ | $\frac{4+6\,\beta+5\,\beta^2+2\,\beta^3}{3+2\,\beta+\beta^2}$ | $1+\beta+2\,\beta^2$ | (5) |

The proof of this Theorem is given in [2]. For $0 \le \beta \le 1/2$ the strategies $\boldsymbol{q}^*$ and $\boldsymbol{p}^*$ are determined such that they render the adversary indifferent with respect to his strategy choice. For $1/2 < \beta \le 1$ one sees immediately that $\boldsymbol{q}^*$ and $\boldsymbol{p}^*$ fulfil the saddle point conditions (2). As already mentioned, these results will be discussed, together with those of the other variants, in the last section of this chapter.

Let us conclude this section with a technical remark: One sees from equations (4) and (5) that

$$\boldsymbol{p}^*\,(1/2) = (1,0,0) \quad \text{and} \quad E^*_{No-No}\,(1/2) = 1+\beta+2\,\beta^2|_{\beta=1/2},$$

which implies that $\boldsymbol{p}^*$ as well as $E^*_{No-No}$ are continuous functions of $\beta$. This, however, is not true for the operator's optimal strategy given by (3). Here we have

$$\boldsymbol{q}^*(1/2) = \left(\frac{6}{17}, \frac{4}{17}, \frac{11}{17}\right) \neq (1,0,0)\,.$$

In fact, the case $\beta = 1/2$ plays a special role: It can be shown, see [2], that for $\beta = 1/2$ the operator has infinitely many optimal strategies given by

$$\boldsymbol{q}^*(\gamma) = \gamma\left(\frac{6}{17}, \frac{4}{17}, \frac{11}{17}\right) + (1-\gamma)(1,0,0)$$

for each $\gamma \in [0,1]$. This implies that

$$\lim_{\beta \to 1/2-} \boldsymbol{q}^*(\beta) = \left(\frac{6}{17}, \frac{4}{17}, \frac{11}{17}\right) \in \left\{\gamma\left(\frac{6}{17}, \frac{4}{17}, \frac{11}{17}\right) + (1-\gamma)(1,0,0) : \gamma \in [0,1]\right\}.$$

This property of the set of optimal strategies is well-known and investigated in detail in [5]. Similar relations also hold for the set of optimal strategies given in Theorem 2 to Theorem 4. However, due to the more technical aspect of these considerations, it will not be mentioned there.

## 2.2. The No-Se Variant

Here the operator behaves in the same way as in the previous variant: He decides at time point $0$ when to start his illegal activity, namely at $0$ or at $1$ or at $2$. The inspector, however, decides at time point $0$ only if to perform the first inspection at time point $1$ or $2$. In case that the first inspection is carried through at time point $1$, he decides if to perform the second inspection at time point $2$ or $3$. In case the first inspection takes place at time point $2$ the second one has to be performed at time point $3$.

The extensive form of this variant is represented graphically in Figure 3. Due to the sequential behaviour of the inspector this extensive form has a more complicated structure than the previous variant.

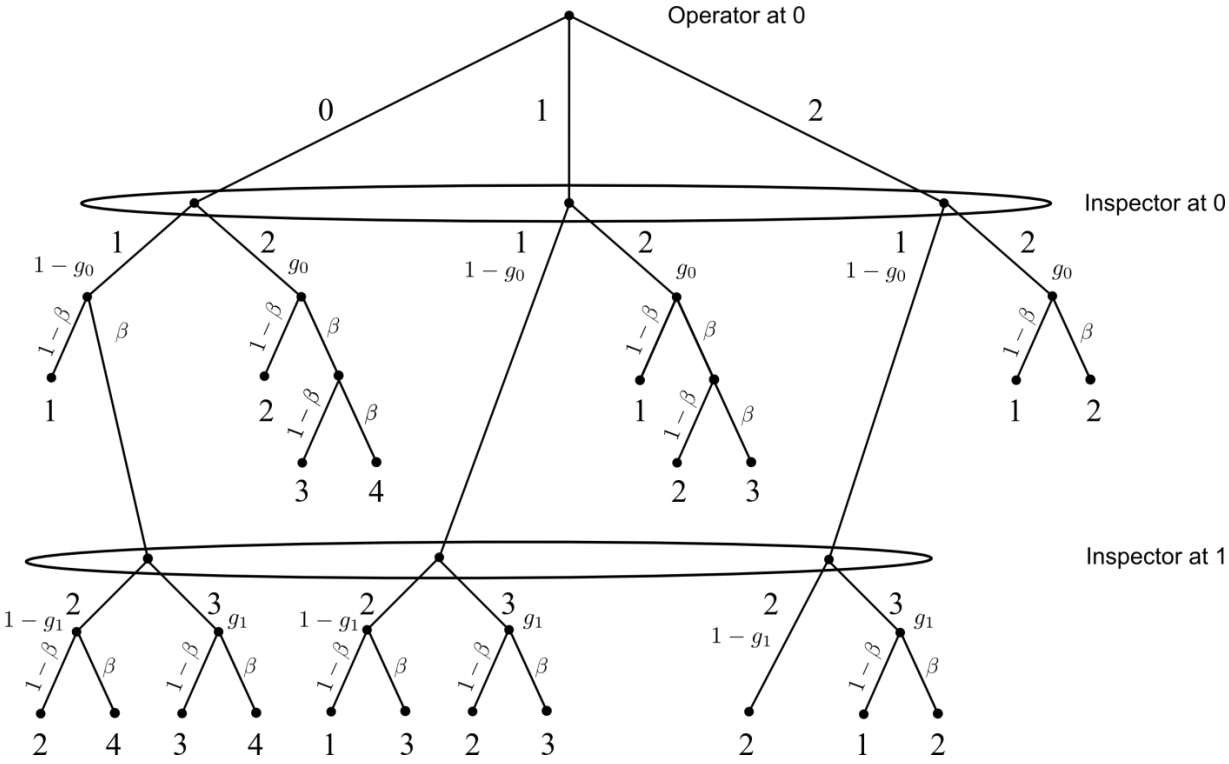


Figure 3: Graphical representation of the extensive form of the No-Se variant of the inspection game. The two encircled areas denote the two information sets of the inspector.

As in the previous variant, let $\boldsymbol{q} = (q_0, q_1, q_2)$ with $q_0 + q_1 + q_2 = 1$ be the operator´s probabilities to choose the time points $0$,$1$ and $2$ for the start of the illegal activity. Furthermore let $1 - g_0$ and $g_0$ be the inspector´s probabilities to perform his first inspection at time point $1$ or $2$, and $1 - g_1$ and $g_1$ the inspector´s probabilities to perform his second inspection at $2$ or $3$ once he performs the first one at $1$. Then the expected payoff to the operator, i.e., the overall expected detection time is

$$\begin{aligned} E_{No-Se}(\boldsymbol{q},\boldsymbol{g}) &= q_0\,[\,(1-g_0)\left(1-\beta+\beta\left((1-g_1)\,(2+2\,\beta)+g_1\,(3+\beta)\right)\right)+g_0\,(2+\beta+\beta^2\,)\,] \\ &+ q_1\,[\,(1-g_0)\left((1-g_1)\,(1+2\,\beta)+g_1\,(2+\beta)\right)+g_0\,(1+\beta+\beta^2\,)\,] \\ &+ q_2\,[\,(1-g_0)\left((1-g_1)\,2+g_1\,(1+\beta)\right)+g_0\,(1+\beta\,)\,]\,. \end{aligned} \tag{6}$$

Again, we are looking for optimal strategies, i.e., a saddle point of this game, which is defined in analogy to (2).

**Theorem 2:** Given the non-cooperative two person zero sum game the extensive form of which is represented graphically in Figure 3. The solution of this game, i.e., the optimal strategies $\boldsymbol{q}^* = (q_0^*, q_1^*, q_2^*)$ and $\boldsymbol{g}^* = (g_0^*, g_1^*)$ and the optimal expected detection time $E_{No-Se}^*$, is given as follows:

| | $0 \le \beta \le 1/2$ | $1/2 < \beta \le 1$ |
|---|---|---|
| $\boldsymbol{q}^*$ | $\frac{1}{3+2\,\beta+\beta^2}\,(1+\beta\,,1\,,1+\beta+\beta^2)$ | $(1,0,0)$ |
| $\boldsymbol{g}^*$ | $\frac{1-2\,\beta}{1-\beta}\;\frac{1+\beta}{3+2\,\beta+\beta^2},\;\frac{(1-2\,\beta)(1+\beta+\beta^2)}{2+2\,\beta-\beta^2}$ | $0,0$ |
| $E_{No-Se}^*$ | $\frac{4+6\,\beta+5\,\beta^2+2\,\beta^3}{3+2\,\beta+\beta^2}$ | $1+\beta+2\,\beta^2$ |

(7)

The proof of this Theorem goes along the same lines as explained after Theorem 1.

With (3) and (5) we see that $\boldsymbol{q}^*$ and $E_{No-Se}^*$ are the same as for the No-No variant even though the structures of the extensive form of both variants are so different. Comparing, however, the expected payoffs (1) and (6) of both variants, one can identify them if one replaces in (1) $p_{(1,2)}$, $p_{(1,3)}$ and $p_{(2,3)}$ by

$$p_{(1,2)} = (1-g_0)\,(1-g_1)\,, \qquad p_{(1,3)} = (1-g_0)\,g_1\,, \qquad p_{(2,3)} = g_0\,. \tag{8}$$

Thus, it is no longer surprising that the two variants lead to the same optimal expected detection times and, of course, the optimal strategies $\boldsymbol{p}^*$ and $\boldsymbol{g}^*$ as given by (4) and (7) reflect these relations.

## 2.3. The Se-No Variant

In this variant, the inspector behaves like in the first variant: without knowledge of what the operator does he decides at time point $0$ when to perform his two inspections, namely at $(1,2)$ or at $(1,3)$ or at $(2,3)$. The operator decides at time point $0$ to start his illegal activity immediately or to postpone it. In the latter case he decides after the first inspection to start his illegal activity immediately or to postpone it again. In the latter case he starts it after the second inspection.

The extensive form of this variant, which also has been analysed already in [2], is represented graphically in Figure 4. Even though one could also represent it in line with the two previous variants, namely starting with the operator´s choices, it seems to be more natural here and in the last variant, to start with the inspector´s choices.

As in the first variant, let $\boldsymbol{p} = (p_{(1,2)}, p_{(1,3)}, p_{(2,3)})$ with $p_{(1,2)} + p_{(1,3)} + p_{(2,3)} = 1$ be the inspector´s probabilities to choose the pairs of time points $(1,2)$, $(1,3)$ and $(2,3)$ for the two inspections. Furthermore, let $1 - h_0$ and $h_0$ be the operator´s probabilities to start the illegal activity right at the

beginning or to postpone it, and in the latter case let $1-h_1$ and $h_1$ be the probabilities to start the illegal activity right after the inspection at time point 1 or to postpone it again.

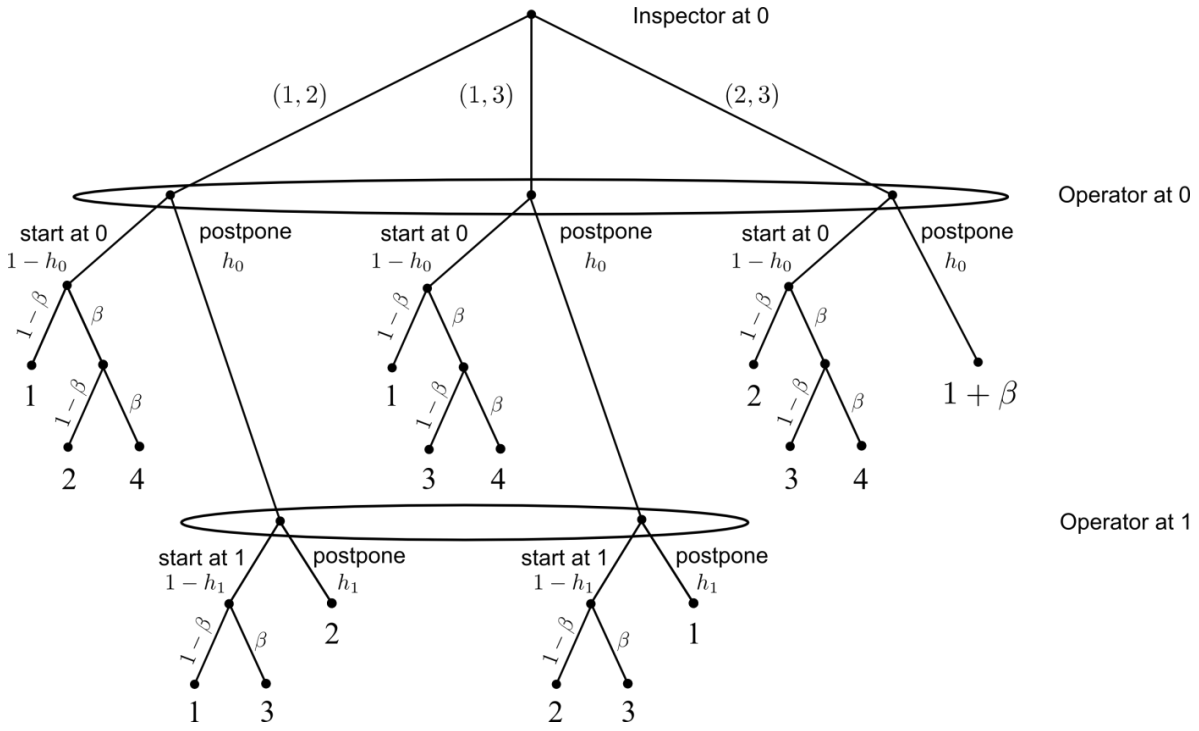


Figure 4: Graphical representation of the extensive form of the Se-No variant of the inspection game. The two encircled areas denote the two information sets of the operator.

Then the expected payoff to the operator, i.e., the overall expected detection time is

$$\begin{aligned} E_{Se-No}(\boldsymbol{h},\boldsymbol{p}) &= p_{(1,2)}\,[\,(1-h_0)\,(1+\beta+2\,\beta^2)+h_0\,(\,(1-h_1)\,(1+2\,\beta)+h_1\,2)\,] \\ &+ p_{(1,3)}\,[\,(1-h_0)\,(1+2\,\beta+\beta^2)+h_0\,(\,(1-h_1)\,(2+\beta)+h_1\,1)\,] \\ &+ p_{(2,3)}\,[\,(1-h_0)\,(2+\beta+\beta^2)+h_0\,(1+\beta)\,]. \end{aligned} \tag{9}$$

Again, we are looking for optimal strategies, i.e., a saddle point of the game, which is defined in analogy to (2). It is given by

**Theorem 3:** Given the non-cooperative two person zero sum game the extensive form of which is represented graphically in Figure 4. The solution of this game, i.e., the optimal strategies $\boldsymbol{h}^* = (h_0^*, h_1^*)$ and $\boldsymbol{p}^* = \left(p_{(1,2)}^*, p_{(1,3)}^*, p_{(2,3)}^*\right)$ and the optimal expected detection time $E_{Se-No}^*$, is given as follows:

| | $0 \le \beta \le 1/2$ | $1/2 < \beta \le 1$ |
|---|---|---|
| $\boldsymbol{h}^*$ | $\dfrac{2-2\,\beta+\beta^2-\beta^3}{3-3\,\beta+2\,\beta^2-\,\beta^3}, \dfrac{1}{2+\beta^2}$ | $1, [0,1]$ |
| $\boldsymbol{p}^*$ | $\dfrac{1}{3-3\,\beta+2\,\beta^2-\,\beta^3}\,(1+\beta+\beta^2+\beta^3\,,\quad 1-2\,\beta+\beta^2-2\,\beta^3,\quad 1-2\,\beta\,)$ | $(1,0,0)$ |
| $E_{Se-No}^*$ | $\dfrac{4-\beta+\beta^2}{3-3\,\beta+2\,\beta^2-\,\beta^3}$ | $1+\beta+2\,\beta^2$ |

(10)

Again, the proof of this Theorem goes along the same lines as explained after Theorem 1.

Contrary to the two previous variants, the optimal expected detection time and all optimal strategies are different from both for $0 < \beta < 1/2$. It can be shown that

$$E^*_{Se-No} > E^*_{No-No} = E^*_{No-Se}$$

holds. This will be discussed in the concluding section of this chapter as well.

## 2.4. The Se-Se Variant

In this fourth and last variant, the inspector behaves like in the second variant: he decides at time point zero only if to perform the first inspection at $1$ or $2$. In case the first inspection is carried through at time point $1$ he decides if to perform the second inspection at time point $2$ or $3$. In case the first inspection takes place at time point $2$ the second one has to be performed at time point $3$. The operator behaves like in the third variant: He decides at time point $0$ to start his illegal activity immediately or to postpone it. In the latter case he decides after the first inspection to start his illegal activity immediately or to postpone it again. In the latter case he starts it after the second inspection.

The extensive form of this fourth and last variant is represented graphically in Figure 5. Compared to the previous variants it has the most complicated structure which would not be simpler if we would start with the operator´s choices at $0$.

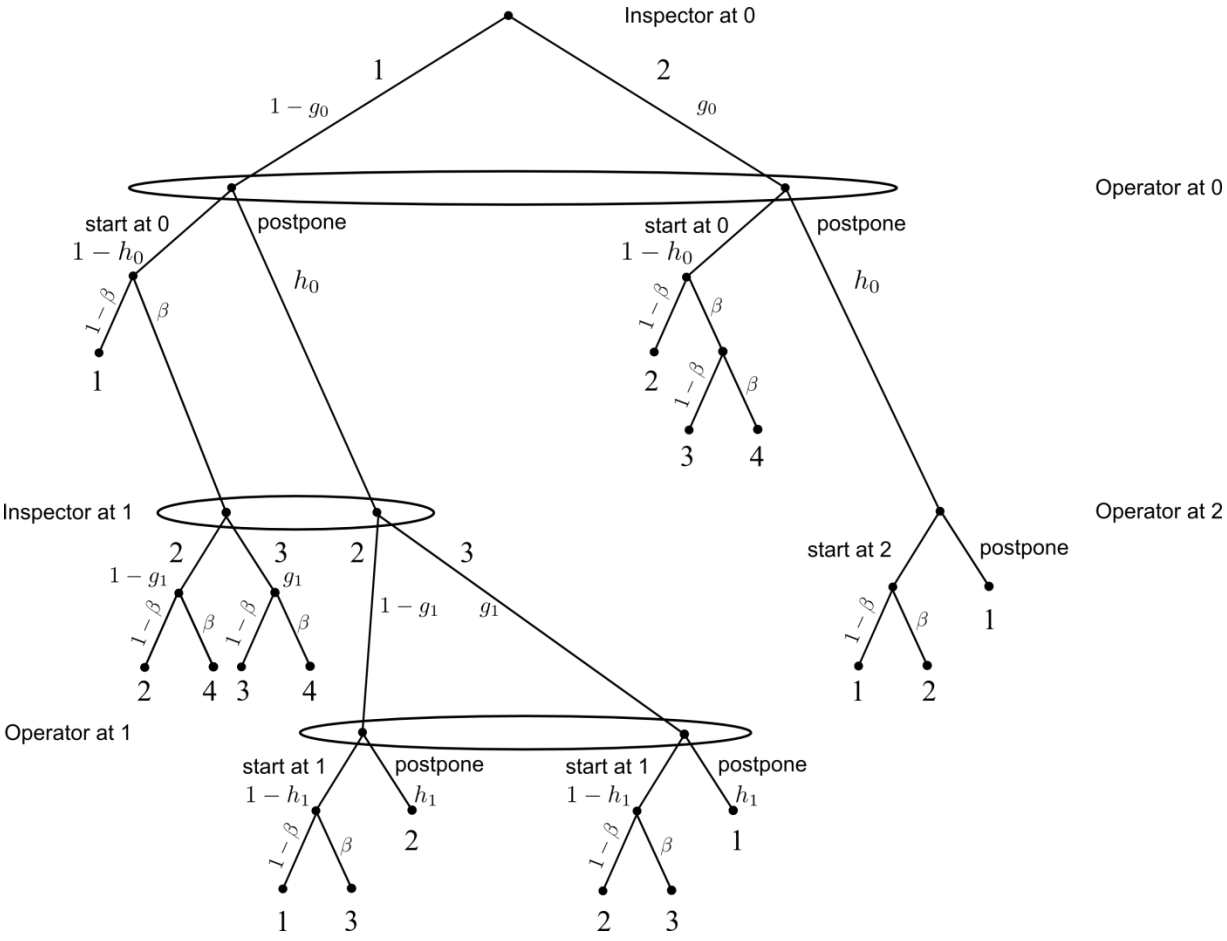


Figure 5: Graphical representation of the extensive form of the Se-Se variant of the inspection game. $\boldsymbol{h}$ and $\boldsymbol{g}$ have the same meanings as in the two previous variants. The three encircled areas denote the two information sets of the operator and the information set of the inspector.

We see that, if the inspector decides at 0 for 2 and the operator at 0 for postponing the illegal activity, then the inspector´s choice for 3 is dominated thus, the operator´s expected payoff is $1-\beta+2\,\beta=1+\beta$. The overall expected payoff to the operator, i.e. the overall expected detection time is given by

$$\begin{aligned} E_{Se-Se}(\boldsymbol{h},\boldsymbol{g}) \;=\; & (1-g_0)\,[\,(1-h_0)\,(1-\beta+\beta\,[\,(1-g_1)\,(2+2\,\beta)+g_1\,(3+\beta)\,]\,) \\ & \quad +h_0\,[\,(1-g_1)\,[\,(1-h_1)\,(1+2\,\beta)+h_1\,2\,]+g_1[\,(1-h_1)\,(2+\beta)+h_1\,1\,]\,]\,] \\ + \;\; & g_0\,[\,(1-h_0)\,(2+\,\beta+\beta^2)+h_0\,(1+\beta\,)\,]\,. \end{aligned} \tag{11}$$

Again, we are looking again for optimal strategies, i.e., a saddle point of the game, which is given in analogy to (2). It is given by

**Theorem 4:** Given the non-cooperative two person zero sum game the extensive form of which is represented graphically in Figure 5. The solution of this game, i.e., the optimal strategies $\boldsymbol{h}^*=(h_0^*,h_1^*)$ and $\boldsymbol{g}^*=(g_0^*,g_1^*)$ and the optimal expected detection time $E_{Se-Se}^*$, is given as follows:

| | $0\le\beta\le 1/2$ | $1/2<\beta\le 1$ |
|---|---|---|
| $\boldsymbol{h}^*$ | $\dfrac{2-2\,\beta+\beta^2-\beta^3}{3-3\,\beta+2\,\beta^2-\,\beta^3},\dfrac{1}{2+\beta^2}$ | 0,0 |
| $\boldsymbol{g}^*$ | $\dfrac{1-2\,\beta}{3-3\,\beta+2\,\beta^2-\,\beta^3},\dfrac{1-2\,\beta}{2-\beta}$ | 0,0 |
| $E_{Se-Se}^*$ | $\dfrac{4-\beta+\beta^2}{3-3\,\beta+2\,\beta^2-\,\beta^3}$ | $1+\beta+2\,\beta^2$ |

(12)

Again, the proof of this Theorem goes along the same lines as explained after Theorem 1.

We see that the optimal strategy $\boldsymbol{h}^*$ of the operator as well as the optimal expected detection time are the same as in the previous variant, see Theorem 3. Thus, we have in total for $0<\beta<1/2$

$$E_{Se-Se}^*=E_{Se-No}^*>\;E_{No-No}^*=E_{No-Se}^*\,.$$

Again, as in the case of Theorem 1 and Theorem 2, we can identify the expected payoffs (9) and (11) of the two variants Se-No and Se-Se, if we replace in (9) $p_{(1,2)}$, $p_{(1,3)}$ and $p_{(2,3)}$ like in (8). And of course again, the optimal strategies $\boldsymbol{p}^*$ and $\boldsymbol{g}^*$ as given by (10) and (12) reflect these relations.

## 3. Discussion of the Results

Let us summarize the most interesting findings in the form of three statements.

First, for $\beta=0$ all four variants lead to the same optimal expected detection time 4/3, i.e., one third of the length of the reference time interval. The same is true for all $1/2<\beta<1$ with the optimal expected detection time $1+\beta+2\,\beta^2$.

Second, the variants No-No and No-Se on one hand, and the variants Se-No and Se-Se on the other, lead for $1/2<\beta<1$ to the same optimal expected detection times. The same holds for the optimal strategies of the operator.

Third, the optimal expected detection times of the two variants No-No and No-Se are smaller than those of the variants Se-No and Se-Se for $0<\beta<1/2$.

In Figure 6 the optimal expected detection times $E^*$ are represented graphically as functions of $\beta$.

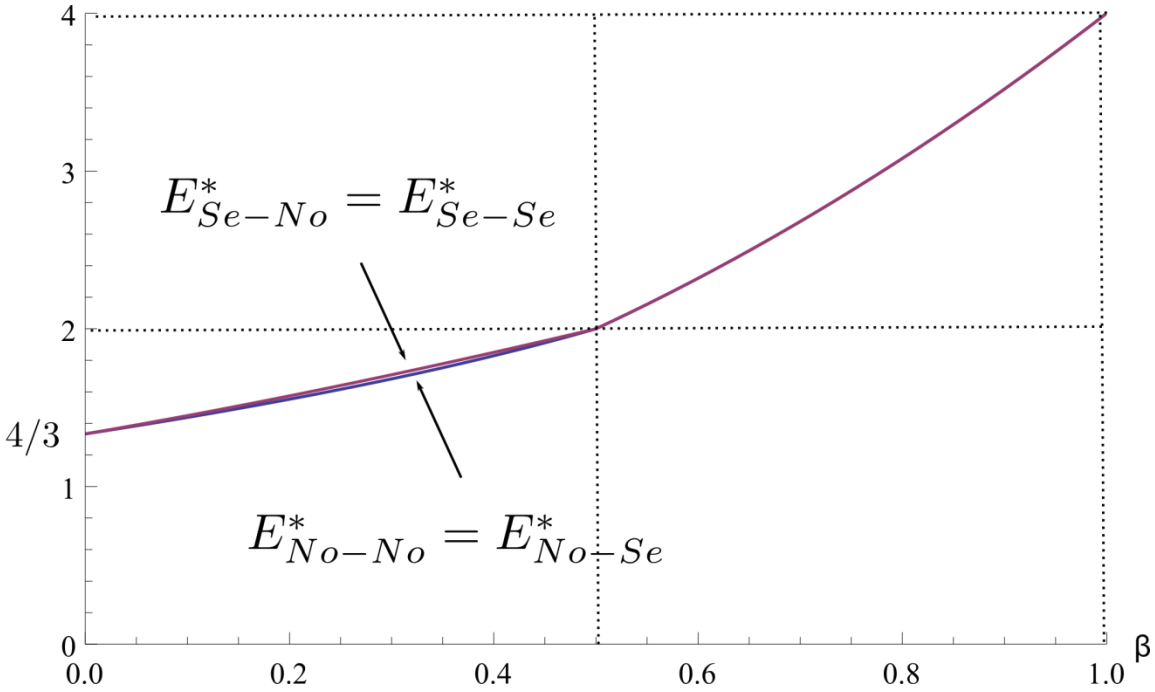


Figure 6: Graphical representation of the optimal expected detection time as function of $\beta$ for all four variants of the inspection game considered here.

In order to find a plausible interpretation of these results one observation is important: There is a distinct difference in information between the two players: if they decide sequentially – independently of what the other one does – the operator knows when an inspection has taken place whereas the inspector, after an inspection which did not prove illegal behaviour of the operator, for $\beta > 0$ does not know if there was no illegal behaviour or else, if he did not detect it. This more comfortable role of the operator may explain the results of the analysis for $0 < \beta < 1/2$; it does not explain, however, why for $1/2 < \beta < 1$ all variants lead to the same optimal expected detection times.

For $\beta = 0$ the inspector knows in case he does not detect during an interim inspection the illegal activity that it has not yet been started. Thus, his information state corresponds in some way to that of the operator. One may doubt, however, if this is a reasonable explanation for the fact that here, i.e., for $\beta = 0$, all four variants lead to the same optimal expected detection time.

## 4. Concluding Remarks

Of course the results of the analysis of the special case of two interim inspections in a nuclear facility at three possible points of time within a reference time interval cannot simply be extrapolated to more complicated situations, i.e., more possible time points and more inspections, even though the information argument given in the previous section holds here as well. On the contrary: Optimal strategies of non-cooperative two person zero sum games exhibit frequently unexpected and surprising properties. But anyhow, if interim inspections have to be planned for more complicated cases than analysed here, and if analytical results for more complicated cases are not available, then one may – with reference to the most simple case – assume that the operator who plans an illegal activity, will behave sequentially, and the inspector may do what it easier for him from an organisational and financial point of view.

Another question arises if interim inspections are possible at *any time* within the reference time interval under consideration, if time is considered to be continuous. Here, three of the four variants have been analysed already: Diamond [6] studied the No-No variant in detail, and Avenhaus, Canty and Krieger [7], [8] both the Se-No and the Se-Se variant. The latter two variants lead again to the same optimal expected detection times. Interesting enough, for two interim inspections and $\beta = 0$ it is

one third of the length of the reference time interval, i.e., the same as obtained here. The analysis of the No-No variant requires a mathematical technique totally different from all other ones, and for two inspections the optimal expected detection time is the fraction $1/(e\,(e-1)) \approx 0.214$ of the length of the reference time interval.

The No-Se variant for continuous time has not yet been studied so far and here, following the experience mentioned above, no prediction concerning the results of the analysis will be attempted. Instead, safeguards experts are invited to continue this work: maybe it can easier be achieved than expected here, but maybe it requires a new mathematical effort comparable to that for the No-No variant.

## 5. Acknowledgement

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The authors gratefully acknowledge the financial support by the Federal Republic of Germany. They are solely responsible for the contents of this work.

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# ULTRA COMPACT HPGe SPECTROMETER FOR IN-SITU MEASUREMENTS

**A.Sokolov[1], E.Loshevich[1], S.Kim[1], J.Brutscher[2], A.Rozite[1], A.Nazarenko[1]**

[1] Baltic Scientific Instruments, Ganibu dambis 26, Riga, LV-1005, Latvia
[2] GBS Elektronik, Bautzner Landstraße 22, 01454 Großerkmannsdorf, Germany
E-mails: a.sokolov@bsi.lv, e.loshevich@bsi.lv, s.kim@bsi.lv,
brutscher@gbs-elektronik.de, a.rozite@bsi.lv, a.nazarenko@bsi.lv

***Abstract:***

*In-situ applications require a very high level of portability of high-resolution spectrometric equipment. Usage of HPGe detectors for radioactivity measurements in the environment or for nuclear safeguard applications, for example, to combat of illicit trafficking of nuclear materials or uranium and plutonium monitoring in nuclear wastes became a norm in the recent years.*
*Recently, portable HPGe-based radionuclide spectrometer with electrical cooling appeared on the market for in-situ applications. At the same time deterioration of energy resolution associated with vibrations produced by cryocooler or high weight of the instrument, short time of autonomous operation and high price of these spectrometers are limiting their usage in many cases.*
*In this paper we present development results of ultra compact hand-held all-in-one spectrometer for in-situ measurements based on HPGe detector cooled by liquid nitrogen without listed disadvantages.*

**Keywords:** HPGe spectrometers, in-situ radioactivity measurements

## 1. Introduction

Usage of HPGe detectors for radioactivity measurements for environmental and nuclear safeguard applications has became a necessity. Different kinds of solutions available to fight against illicit trafficking of nuclear materials or uranium and plutonium monitoring in nuclear wastes. Having excellent energy resolution and high intrinsic registration efficiency HPGe detectors have identification accuracy advantages over spectrometric scintillation and room temperature compound semiconductor detectors. However, HPGe detectors have to be cooled down to temperature below 100 K [1].

Ultra compact hand-held HPGe detector cooled by liquid nitrogen has been developed by us earlier for in-situ applications [2]. The instrument was equipped with planar HPGe detector with sensitive area of 500 $mm^2$. Its performance was showing excellent energy resolution, better than 580 eV at 122 keV, extremely short cool down time, less than 1.5 hours, long holging time, more than 20 hours and low weight, less than 2.6 kg.

Hand-held detector was combined with a small multi-channel analyzer [3] powered by battery and with mini-laptop for specra processing and visualization of the measurement results. Despite compact design of the all system components, application of the solution in the field is inconvenient due to nescessity of carrying of all components separately.

Recently, portable HPGe-based radionuclide identifiers/spectrometers with electrical cooling for in-situ applications [4, 5] appeared on the market. Despite many advantages, such solution still have several disadvantages like deterioration of energy resolution associated with vibrations produced by cryo-cooler, high weight of the instrument, short time of autonomous operation and high price.
Current document presents research and development result of ultra compact hand-held all-in-one spectrometer for in-situ measurements based on HPGe detector cooled by liquid nitrogen. A newly developed system solves all mentioned above disadvantages. Upgraded detection, cooling and spectrometric parts were implemented.

## 2. Spectrometer Design

Hand-held HPGe spectrometer was designed and developed to fulfil requirements being placed on radioisotope identifiers [6] and was called NitroSPEC (Fig.1).

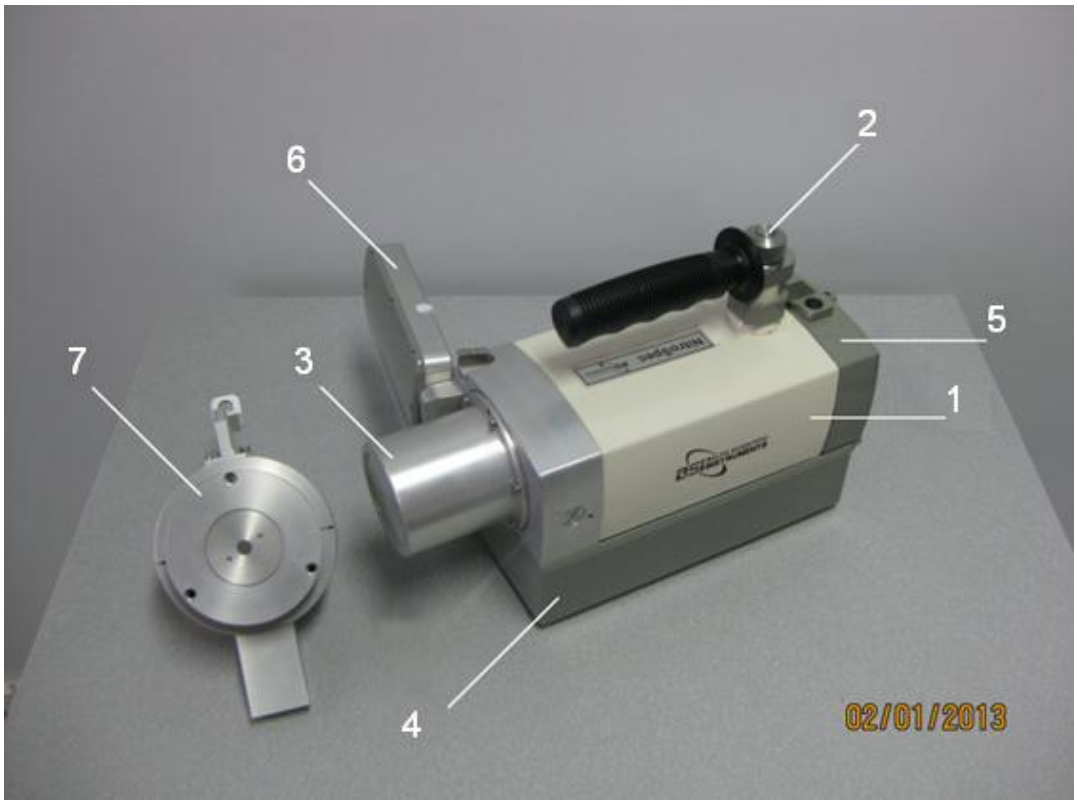


*Fig.1. Design of all-in- one hand-held HPGe spectrometer NitroSPEC.*
*1 –Dewar vessel; 2 –liquid nitrogen filling valve; 3 –cryostat with HPGe detector;*
*4 –electronics section; 5 –power supply section; 6 –touch screen display;*
*7 -removable shielding cap with collimator.*

Spectrometer cooling system is based on the Dewar vessel (1) of the same design as in [2], but its volume was enlarged from 0.6 to 0.8 litres to provide possibility to install HPGe detectors of larger volume. Dewar vessel could be fully filled with liquid nitrogen in less than 10 minutes by means of filling funnel at normal pressure. To keep dimensions of spectrometer as small as possible Dewar vessel has one valve (2) for liquid nitrogen filling and nitrogen evaporation. Depending on the location of valve spectrometer could be used in horizontal and upright or in horizontal and downright orientation.

Dewar vessel volume enlargement allows installing in cryostat (3) coaxial HPGe detectors with efficiency up to 20% and planar detectors with sensitive area up to 2000 $mm^2$ [7]. Nevertheless, the instrument still has short cooling time of the HPGe detector within 1.5 hours after liquid nitrogen filling and provides 24 hours of autonomous non-stop operation time between liquid nitrogen refilling. Refilling of liquid nitrogen once a day is convenient enough for routine measurements on day-by-day basis.
All electronic components are hermetically sealed in a separate section (4) below the Dewar vessel. Hermetic section for batteries (5) is located behind Dewar vessel. Display (6) is made as folding unit.

Some applications require detector to be shielded from external interference. Initially 10 mm thick lead cap and collimators of different diameters were developed. 7 mm thick tungsten alloy cap having smaller dimensions with removable collimators have been developed and fabricated especially for uranium enrichment measurements. Caps (7) could be easily attached to the Dewar vessel flange through special holes by means of 3 tungsten screws. Collimators have different diameters: 40, 25, 10 and 5 mm and could be screwed into a cap. In order to decrease X-ray fluorescence from tungsten, cap and collimators have 1 mm thick tin internal lining covered with 1.5 mm thick copper lining.

Dimensions of the spectrometer are only 330x140x210 mm. Total weight of the spectrometer based on 20 % efficiency HPGe detector without liquid nitrogen is 4.950 kg.

## 3. Electronics and software

Preamplifier with resistive feedback is hermetically sealed in a separate section below the Dewar vessel along with digital signal processing (DSP) electronics (16 k) [8] with integrated high voltage power supply for detector (up to ±3.6 kV) and preamplifier power supply (±12 V, 60 mA). Linux based miniature PC allows to control all operation modes of the spectrometer, electronic health status diagnostic as well as transfer of the accumulated spectra. The built-in microprocessor has nonvolatile memory up to 32 Gb, what allows storage of practically unlimited acquired spectra and the results of their procession.

All spectrometer settings could be set in the menu. Spectrum acquisition could be made in live time and real time modes as well as without time limitation. Practically unlimited extension of nuclides library base is possible as well as the loading of a new measurement geometries including simulated by Monte Carlo method.

With a help of folding built-in display, the user is able to make precise analysis and obtain results in on-line mode with automatic GPS position data indication.

Visualization of measurement results and control of spectrometer parameters is made by means of color built-in touch screen display having resolution of 800x400 pixels and diagonal 4”. Applied SuperAmoled display provides high brightness and contrast of image, what provide comfort operation even at bright sun light. Complex calculations in real time and results display are provided by high frequency (1 GHz) processor and video accelerator.

The connection to another PC for data exchange could be made by USB connection or by wireless WiFi network. Portable spectrometer is fully supported and controlled by expert software of SpectraLine family [9], which provides its application with facilities of precision laboratory spectrometry.

In **easy mode** of spectrometer operation user can select continuous or preset time of measurement and define desirable time interval in seconds. Continuous measurement of spectrum starts by pushing only a single button. Information is updated every second (Fig.2). Manual energy calibration from touch screen is performed selecting automatically identified spectral lines with count rates above background level and setting corresponding energies using library of radionuclides.

In **search mode** of operation a Dose rate is calculated from integral spectrum and is visualized on a display in μSv/h. In case of excess of dose rate over user defined threshold instrument produce visual and acoustic alarm. Identification of radionuclides is performed according to library of radionuclides automatically. List of identified radionuclides and corresponding count rates appears on the screen. Parameters of spectrometric device are easily adjustable from touch screen (Fig. 3).

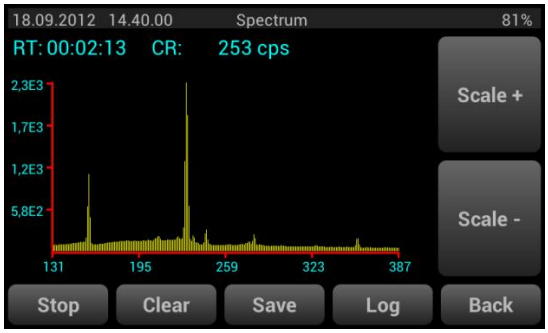


*Fig. 2. Spectrum window available from touchscreen of NitroSPEC spectrometer.*

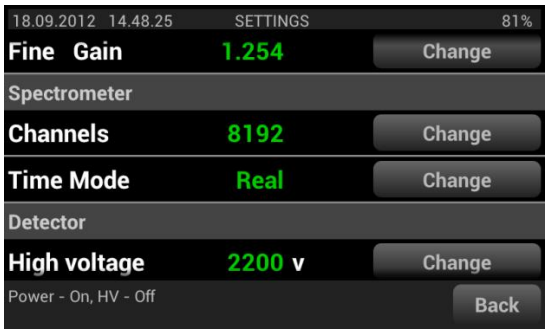


*Fig. 3. MCA settings window available from touchscreen of NitroSPEC spectrometer.*

In **expert mode** of operation user can select region of interest or ratio of regions of interest on the spectrum. This option is convenient for example for U enrichment measurements in UF6 cylinders. User could select type of spectra acquisition between real time, live time or integral (the measurement will continue until a certain net area of a peak is reached) as well as select repeated acquisition mode of measurements. Spectrometer can be connected in parallel to laptop via wireless data transfer interface in expert mode of operation. This option is convenient for calculation of activities of radionuclides.

Additionally spectrometer equipped with two batteries (type Li-Ion) providing power supply of all spectrometer electronic during more than 8 hours without re-charging. Field replacement (hot swapping) of batteries to achieve 16 hours continuous operation time is possible. When accumulators are discharged the software will issue appropriate message. If it is ignored, HV will be shut down automatically when critical low battery charge is achieved to prevent possible damage of the detector.

## 5. Conclusion

Despite that presented spectrometer requires liquid nitrogen for operation, this instrument is well suited for such in-situ applications as border monitoring of illicit trafficking of radioactive materials or nuclear safeguards measurements. Prevention of illicit trafficking of nuclear materials requires reliable and fast radionuclides identification. This is guaranteed by the given spectrometer by means of excellent parameters of energy resolution and high intrinsic registration efficiency of HPGe detector.
Visualisation of the measurement results and identification of radionuclides is realized in the current spectrometer on the basis of automated software algorithms. Control of spectrometer parameters and data evaluation is available for operator from a color built-in touch screen display in immediate vicinity to HPGe detector.

For those nuclear safeguards applications where optimal energy resolution at low energies is of advantage, for example U and Pu monitoring in nuclear wastes, application of liquid nitrogen cooled HPGe detectors is still more effective then electrically cooled because of the negative impact of microphonic effects on the energy resolution of the later ones.
In the mean time possibility to avoid both limitations i.e. usage of liquid nitrogen as well as avoid vibrations produced by cryocooler, maintaining compact design of the instruments intended for in-situ measurements, is still a good perspective for future developments in the field.

Dimensions of developed spectrometer are only 330x140x210 mm, maximum weight without liquid nitrogen is 4.950 kg (with 20% efficiency detector) and time of autonomous operation is at least 8 hours before charging or replacement of batteries.

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# Sensitivity of the neutronic design of an Accelerator-Driven System (ADS) to the anisotropy of yield of the neutron generator and variation of nuclear data libraries

**Petty Cartemo[1], Anders Nordlund[1], Dina Chernikova[1], Wang Ziguan[1]**

[1]Chalmers University of Technology, Department of Applied Physics, Nuclear Engineering, Fysikgarden 4, SE-412 96 Gothenburg, Sweden
E-mail: petty@nephy.chalmers.se

***Abstract:***

*The Accelerator-Driven System concept was chosen to be a basis for the Multi-purpose Hybrid Research Reactor for High-tech Applications (MYRRHA), which can operate in both sub-critical and critical mode. Therefore, the design studies in the scope of this project were varying from accelerator and material aspects to the demonstration of transmutation possibilities of the system. However, the sensitivity of neutron characteristics of the system to the anisotropy of yield of neutron generator and variation of nuclear data libraries may appear to be an important issue.*

*In this study the corresponding sensitivity analysis was performed in order to evaluate the sensitivity of ADS neutronic design to the variation of nuclear data libraries and influence of the anisotropy of the neutron yield of the accelerator.*

**Keywords:** ADS; lead; neutron generator; neutron yield; data libraries

## 1. Introduction

Nowadays, Accelerator Driven Systems (ADS) are considered to be very promising and attractive nuclear systems because of their potential for transmutation of long-lived fission products and superior safety characteristics. Therefore, there are a number of countries around the world which strongly support programs for research and development of ADS.

It is worth to mention, that the significant part of the research program is related to the optimization of the ADS design and safety characteristics by using Monte-Carlo simulations. Therefore, it is very important to have a reliable neutronic design of ADS and be aware of uncertainties and inaccuracies which can appear in the modelling process.

The typical design of ADS includes a subcritical fast spectrum core surrounded by lead reflector and coupled with a particle accelerator. As one can see, there are a number of components which could cause the difficulties, such as uncertainties, inaccuracies and even misinterpretation of results of numerical modelling of the ADS systems. Among them, we can highlight the factors related to the use of different data libraries for lead isotopes, as well as, possible influence of the anisotropy of yield of neutron generator to neutron characteristics of the fast spectrum system.

As it was shown earlier in [1], large differences are observed in the slowing-down time characteristics of neutrons due to the variation of nuclear data libraries of lead.

Some studies also indicate that the anisotropy of the neutron yield of the accelerator should be carefully accounted in both experiments and numerical modelling of the fast spectrum systems [2]. Since ADS systems generally use a fast spectrum core the evaluation of this factor appears to be an important issue.

Therefore, in this study we evaluate the influence of the anisotropy of the neutron yield

of the accelerator and different data libraries to the different characteristics of ADS system.

## 2. The set-up for numerical simulations

The geometry that was used for the simulation of the problem corresponds to the VENUS-F core which is described in [3, 4]. In Fig. 1, positions of neutron source and detectors are shown. Twenty point detectors were used in the same experimental channel with constant x-y coordinates (at x = 8.035; y = -16.135) [4], whereas the z-coordinate varies with irregular intervals between 49 cm to -110 cm.

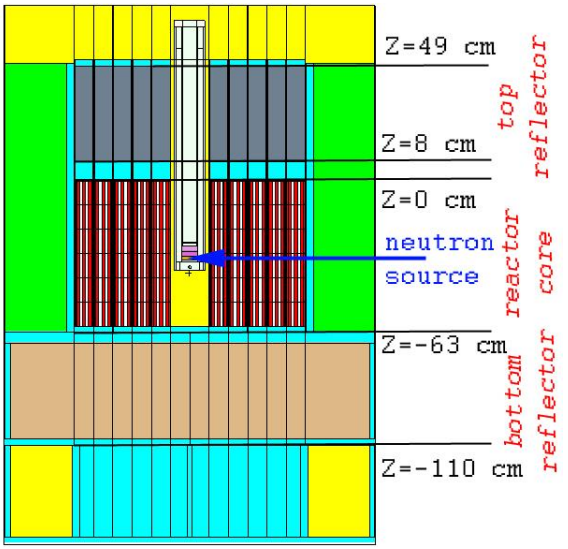


**Figure 1:** a simplified illustration of the vertical cut through the VENUS-F reactor.

This geometry was used as the basis for the MCNPX and MCNP-PoliMi [5] model configuration. The main difference between the standard MCNP and an MCNP-PoliMi, a modification of MCNP-4C, programs is that the first one simulates correctly mean values of the desired quantities while the second one takes into account the correlation between neutron interaction types and secondary gamma generation. Thus, the output of MCNP-PoliMi can be processed in order to get information on neutron and gamma collisions in the detectors.

By investigating the neutron flux per energy group in different positions along the z-direction of the core, we will demonstrate the influence of the source's anisotropy as well as the influence of data libraries. The neutron source is positioned within the core according to Fig. 1.

Also, the inwards and outwards fluxes through different surfaces are calculated giving the opportunity to investigate the change in albedo, the reflective index of a material.

Albedo is defined as

$$\beta = \phi_+/\phi_-$$

where $\phi_+$ represents the flux coming from the source and $\phi_-$ - the reflected flux from the medium above/below the surface. The values of albedo were calculated in positions of nine various surfaces along the z-direction of the core for neutrons with different energy (up to few MeV). The surfaces' positions represent transitions between different materials within the core, as follows: 50.85 cm, 47.85 cm, 7.95 cm, 0.45 cm, -60.94 cm, -63.16 cm, -68 cm, -108 cm.

### 2.1. Data libraries

There are a number of various data libraries for lead. Most of the cross-section varies for the same lead isotope with respect to origin and version of data libraries. However, major libraries, such as ENDF/B and JEFF are very similar in general. Though, as you can see in Fig. 2 – Fig. 4 below, there are differences observed between the ENDF/B-VII and ENDF/B-VI version for the lead isotopes: $^{207}$Pb, $^{208}$Pb. Similar results can be found for the isotopes of lead: $^{204}$Pb, $^{206}$Pb. Most of the differences are related to the neutron capture cross-section and (n, 2n)-reactions. Both these processes play an important role in the neutron transport and multiplication in a fast spectrum core.

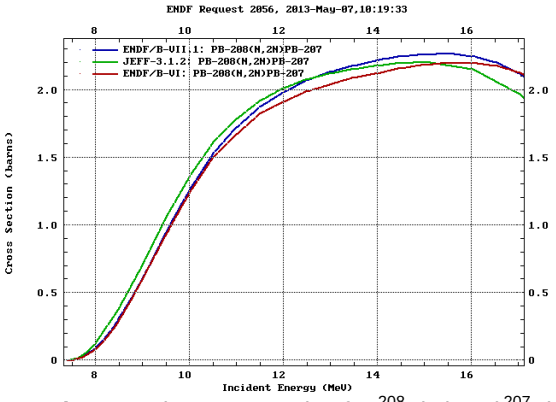


**Figure 2:** the cross section for $^{208}$Pb (n,2n)$^{207}$Pb reaction for ENDF/B-VII, JEFF-3.1.2. and ENDF/B-VI data libraries [6].

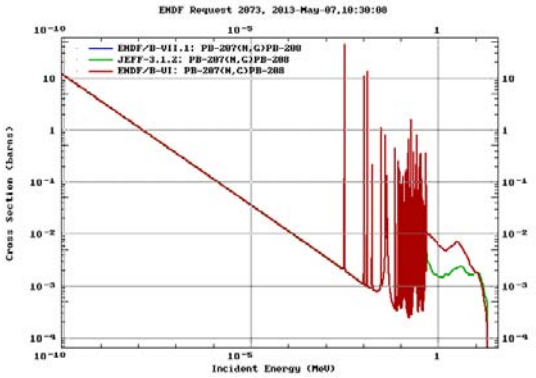


**Figure 3:** the cross section for $^{207}$Pb (n,g)$^{208}$Pb reaction for ENDF/B-VII, JEFF-3.1.2. and ENDF/B-VI data libraries [6].

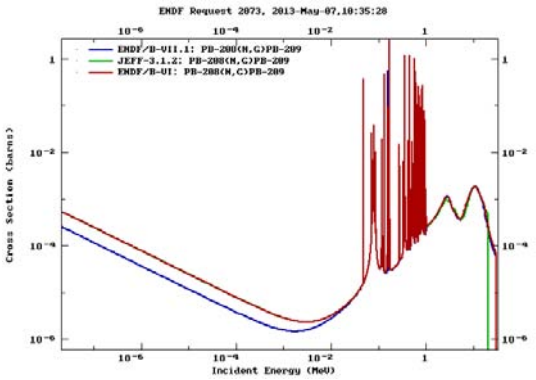


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**Figure 4:** the cross section for $^{208}$Pb (n,g)$^{209}$Pb reaction for ENDF/B-VII, JEFF-3.1.2. and ENDF/B-VI data libraries [6].

Thus, in the present work we investigate the influence of the use ENDF/B-VII data library to the characteristics of ADS system compared to the use of ENDF/B-VI data library. It is necessary to mention that ENDF/B-VII data library is in most cases identical to the JEFF-3.1.2 for lead isotopes with exception of the cross section for $^{208}$Pb (n,g)$^{209}$Pb reaction.

## 2.1. The anisotropy of the neutron yield of the accelerator

The MCNP cannot model the deuteron transport and neutron emission from the D-T reaction for incident deuterons with energies in the keV range (it is only possible for deuterons with energy more than 1 MeV). In order to perform simulation of the D-T neutron production one can use the SOURCE and SRCDX subroutines (after compilation with the MCNP source code) [7] or to calculate the relative neutron yield separately in order to describe the neutron source in MCNP input file. In our research we will use second method. Detailed calculations and tests of the real anisotropic source can be found in [8].

In Fig. 5 one can see the plot of the relative yield of neutrons with particular energy at particular angle in the laboratory (LAB) system, into which the neutron is emitted.

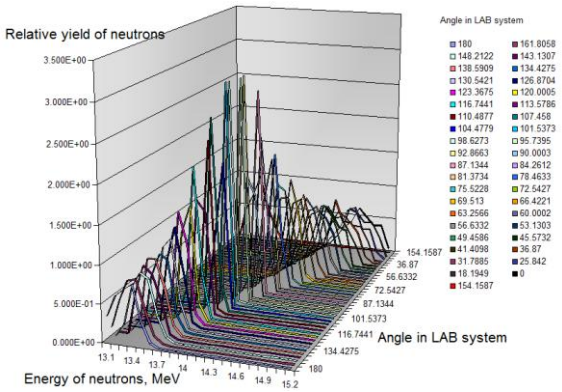


**Figure 5:** the relative yield of neutrons from TiT-target with particular energy at particular angle in the LAB system, into which the neutron is emitted.

## 3. Results and discussion

As a first step, the evaluation of the time-independent characteristics was performed in connection with the use of two different sources, i.e. a real anisotropic source and a point isotropic source with mono-energetic 14 MeV neutrons. Results of the simulation of the neutron flux for 20 different positions within the core are shown in Fig. 6.

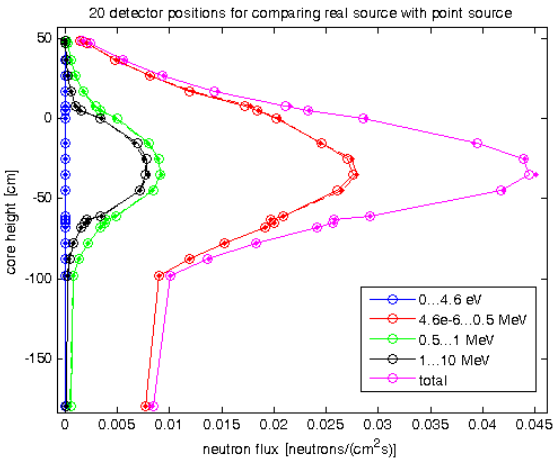


**Figure 6:** the comparison of the flux of neutrons (different energy intervals) for 20 different positions within the core for two cases: the real anisotropic source and the point isotropic source with neutron energy of 14 MeV (dash line corresponds to the results obtained when using the real anisotropic source).

As one can see in Fig. 6, no difference is observed between these two cases for neutrons with energy higher than 0.5 MeV and for thermal energies. Only a slight difference can be seen for neutrons with energy between 0.46 eV and 0.5 MeV when the detector is placed in position Z=(30-50) cm, i.e. nearby the source.

These results are in good agreement with albedo calculations. As one can see from Fig. 7, the overall average neutron reflection coefficients are not changing for different reflectors due to the use of the real anisotropic source and the point isotropic source.

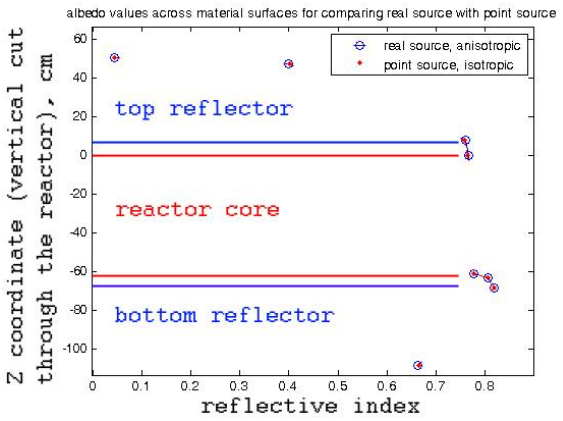


**Figure 7:** the comparison of the values of albedo across different material surfaces along the core height for two cases: the real anisotropic source and the point isotropic source with energy 14 MeV.

Let us now consider time-dependent characteristics of the detector counts, for example, in the core and the reflector. As it is shown in Fig. 8, the time behaviour of neutrons in the detector (placed in core) is very similar for both cases, when the real anisotropic source and the point isotropic source are used. However, in the time interval between 1-100 ns, the total number of counts will be slightly lower for the case when real anisotropic source is used.

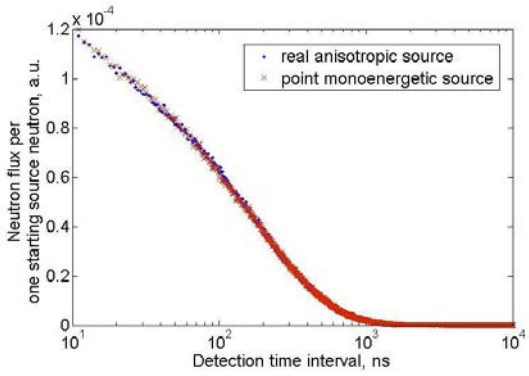


**Figure 8:** the comparison of the time behaviour of neutrons in the detector (placed in the core) for two cases: the real anisotropic source and the point isotropic source with energy 14 MeV.

A similar situation is observed when the detector is placed in the reflector (Fig. 9).

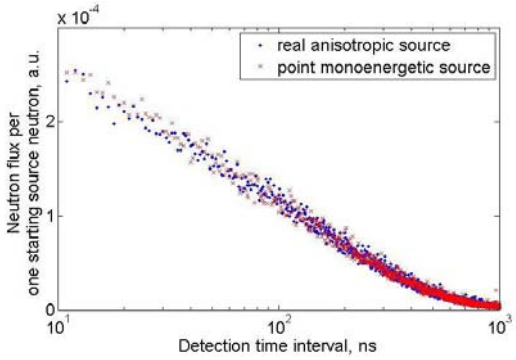


**Figure 9:** the comparison of the time behaviour of neutrons in the detector (placed in the reflector) for two cases: the real anisotropic source and the point isotropic source with energy 14 MeV.

Due to these small differences in the time-dependent characteristics we can also notice the slight difference in the magnitude of the plateau in the Feynman-alpha curve (see Fig. 10).

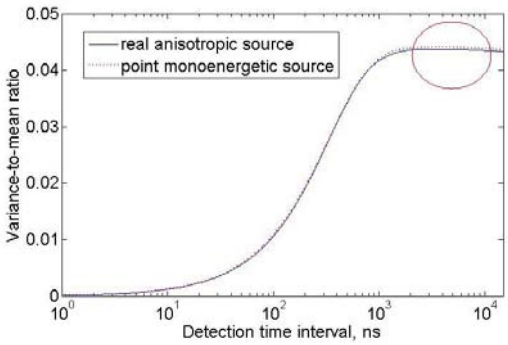


**Figure 10:** the variance to mean ratio (the detector placed in the reflector) for two cases: the real anisotropic source and the point isotropic source with energy 14 MeV.

Thus, we can conclude that the difference between the using of the real anisotropic source and the point isotropic source with 14 MeV neutrons is negligible, except when it comes to simulation time. The MCNPX simulations performed with the point isotropic source took approximately 30% less time. However, before drawing final conclusions, extra simulations in the radial directions are needed when the detectors are placed in positions Z=(30-50) cm, i.e. nearby the source.

The simulations performed for evaluating the influence of the anisotropy of the neutron yield of accelerator were done with a use of ENDF/B-VII data library, which is identical to the JEFF-3.1.2 data library for all lead isotopes with

exception of the cross section for $^{208}$Pb (n,g)$^{209}$Pb reaction.

The next step of this study was connected to the evaluation of the sensitivity of the characteristics of the ADS system to the use of ENDF/B-VI data library for all isotopes of lead, except of $^{204}$Pb, for which the ENDF/B-VII data library was used.

As one can see in Fig. 11, a change of the cross-sections for only lead isotopes develops into significant difference between the fluxes of neutrons of different energy in all 20 different positions of detectors within the core.

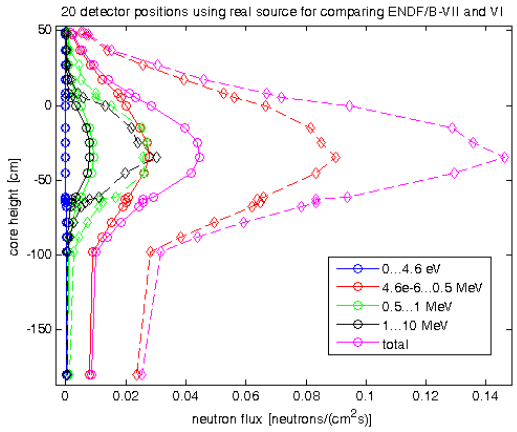


**Figure 11:** the comparison of the flux of neutrons (different energy intervals) for 20 different positions within the core for two cases: ENDF/B-VI and ENDF/B-VII data libraries (dash line corresponds to the results obtained when using ENDF/B-VI data library).

These small differences also affect the time-dependent characteristics, such as variance to mean ratio, as one can see in Fig. 12. In the case of using ENDF/B-VI data library for lead isotopes, this ration is an order of magnitude higher than for the case when ENDF/B-VII data library is used.

However, the changes in the data libraries do not influence to the albedo values calculated across different surfaces along the core height, as one can see in Fig. 13.

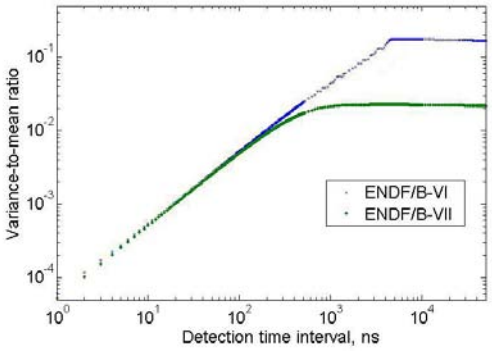


**Figure 12:** the variance to mean ratio (the detector placed in the core) for two cases when ENDF/B-VI and ENDF/B-VII data libraries are used.

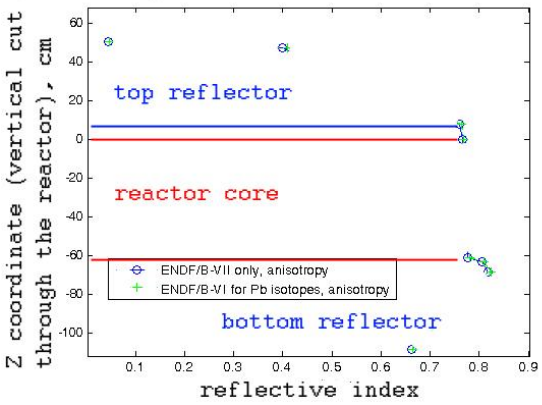


**Figure 13:** the comparison of the values of albedo across different material surfaces along the core height for two cases when ENDF/B-VI and ENDF/B-VII data libraries are used.

## 5. Conclusions

In the present study we have evaluated the influence of the anisotropy of the neutron yield of the accelerator and different data libraries to the different characteristics of ADS system.

Based on the results of the study we can conclude that the difference between using the real anisotropic source and the point isotropic source with energy 14 MeV is negligible except when it comes to simulation time. The MCNPX simulations performed with a point isotropic source with energy 14 MeV took approximately 30% less time. However, for the case when the real anisotropic source is used, in the time interval between 1-100 ns, the total number of counts will be slightly underestimated compared to the same value for the case when the point isotropic source is used. Also, if using two various descriptions of the sources, the slight difference can be seen for neutrons with energy from 0.46 eV to 0.5 MeV when the detector is

placed in position Z=(30-50) cm, i.e. nearby the source. Thus, before coming to final conclusions, additional simulations in the radial directions are needed when the detector is placed in positions Z=(30-50) cm, i.e. nearby the source.

It should also be mentioned that the results of changes of the cross-sections for only the lead isotopes developed into a big difference between the fluxes of neutrons which belong to the different energy intervals for 20 different positions of detectors within the core. However, the changes in the data libraries do not affect the albedo values calculated across different surfaces along the core height.

## Acknowledgement

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# A change detection tool for time series of SAR images

Ph. Loreaux and G. Quin
CEA, DAM, DIF, F–91297, Arpajon, France
philippe.loreaux@cea.fr, guillaume.quin@cea.fr

MIMOSA (Method for Generalized Ordered Means Series Analysis) is a tool dedicated to automatic change detection in time series of SAR images. The method has three distinct modules, respectively dedicated to image pairs, triplets and time series. In each case, the only parameter is the false alarm rate (typically a few %). To detect changes, MIMOSA calculates two different temporal means (geometric and arithmetic mean, for example) from the images, and predicts their behavior if no change had taken place using a rigorous model of amplitude signal. Changed areas are then detected when they do not respect the stable behavior predicted by the model of no change. MIMOSA method does not require image pre-filtering and exploit them in full resolution. Additional change classification tools are also proposed in the case of time series, in order to assist interpretation. A graphical user interface has also been developed and integrated in the software ENVI, MIMOSA then forms a complete chain of change detection and site monitoring. The method shows good results in satellite imagery, as well as airborne images.

# Modernization of EURATOM's unattended measurement stations at Melox MOX Fuel Fabrication and La Hague Reprocessing Plants

**D. Ancius[1], R. Berndt[2], P. Mortreau[2], J. Löschner[3], P. de Baere[1], J.F. Levert[1], W. Eestermans[1], P. Schwalbach[1]**

[1]Nuclear Safeguards Directorate, DG Energy, European Commission,
L-2920 Luxembourg, Luxembourg
[2]ITU, Joint Research Centre, European Commission, Ispra, Italy
[3]IPSC, Joint Research Centre, European Commission, Ispra, Italy
e-mail: Darius.Ancius@ec.europa.eu

***Abstract:***

*The Nuclear Safeguards Directorate performs rigorous controls of the bulk handling nuclear materials sites. The inspectors are present from 25 to 50 percent of the working days in the Melox MOX Fuel Fabrication Plant and the La Hague Reprocessing Plant. The installed independent EURATOM measurement stations assure continuous measurements allowing the inspectors to verify 100 percent of the nuclear material flow across critical points in those facilities. For security reasons, access to these measurement points is usually very restricted. The reliability and high availability of these measurement stations is thus of high significance. Some of them were continuously operating for over 20 years without major maintenance before the upgrade discussed in this paper.*

*Six unattended measurement stations, consisting of neutron coincidence collars and electro-mechanically cooled high purity Germanium detectors have been modernized recently. The reliability of the automated data acquisition system has been significantly improved. The availability of the data transmission networks was improved at the same time. The number of interventions as well as the historically high maintenance costs for the measurement stations is expected to be much lower in the future.*

*This paper describes a number of complex problems that needed to be solved, including the successful technical solutions used for the reduction of EMI (Electro-Magnetic Interference). An overview with respect to the ageing problems of the components of measurement stations discovered during their in depth maintenance will be provided.*

**Keywords:** installed equipment, stores, maintenance, measurement system

## 1. Introduction

Article 77 of the EURATOM treaty stipulates that the European Commission shall satisfy itself that no nuclear material is diverted from the intended use as declared by the users. Therefore, nuclear operators within the European Union are obliged to have a system of accountancy and control for nuclear materials and to inform the European Commission of any inventory changes. Inspectors are being sent to nuclear installations regularly to ensure compliance between physical reality and these declarations.

Because of the high strategic value of plutonium as a direct use material, associated with the difficulty to run thorough physical inventory verification in a very large store, the Nuclear Safeguards Directorate has chosen the principle of continuity of knowledge to plutonium stores as inspection approach. If the complete flow in and out of a store is monitored, the inventory is known at all times. This requires that every move of nuclear material in or out the store must be observed; every item measured, analysed and compared with the operators' declaration. Each plutonium container is measured when it enters the store and again when the container exits the store.

Secondly, the principle of continuity of knowledge implies that no other moves in or out of the store are possible without being observed. Practically at the La Hague and Marcoule nuclear sites (both in France), implementation of this approach is achieved by using a combination of cameras, neutron monitors, seals and unattended measurement stations.

This approach allows the Directorate to optimize the inspectors' presence in UP2 and UP3 Reprocessing Plants (La Hague nuclear site) and in Melox MOX Fuel Fabrication Plant (Marcoule nuclear site) to about 2 routine inspections per month and 2 to 3 inventory inspections per year.

In the above mentioned inspection schemes, the EURATOM measurement stations play thus a key role. Failure of a measurement station (both gamma and neutron) will have a serious impact on the inspection activities.

## 2. Status of EURATOM unattended measurement stations before modernization

In this paper, we discuss EURATOM's unattended measurement stations installed in key points of the La Hague and Melox plants. In principle, the number of measurement stations in a facility depends on structure, mode of operation, material types and flows of the facility and is defined as part of the safeguards approach. In our case, the stations all consist of a neutron coincidence collar and a low energy high purity Germanium detector used to determine the plutonium mass and isotopic composition in the items. This measurement setup permits EURATOM inspectors to evaluate the mass of plutonium independently. The measurement stations are operated in unattended mode by EURATOM data acquisition and evaluation system (RADAR/CRISP), 'Schwalbach [1, 2]. The unattended acquisition system assures automated collection of raw data, separation of measurement events from the background radiation, evaluation and storage of the data in dedicated databases, and it is independent from the operator. The measurement stations discussed here are only a part of the safeguards equipment installed in the two facilities. The information provided by the neutron-gamma stations is combined with many other qualitative and quantitative sensors and surveillance and containment equipment, which are all applied to implement the safeguards approach. The measurement data is automatically transmitted to the local EURATOM office through independent communication lines.

### 2.1. Melox MOX Fuel Fabrication Plant

The original safeguards approach for Melox is in detail explained by 'Arenz [3]. Here we discuss the modernisation of two specific measurement stations which have been installed in the Melox MOX Fuel Fabrication Plant from the beginning. They were in continuous operation for more than 15 years. One station, consisting of a neutron coincidence collar and an electro-mechanically cooled high purity Germanium detector was installed at the exit of the plutonium oxide store (D1). The second station (D2) is a custom design neutron coincidence collar combined with the same kind of gamma detector, 'De Baere [4]. Station D2 measures the MOX fuel rod trays and is located in a rod processing area of the plant. In order to measure plutonium oxide which enters the store, more recently a special design neutron coincidence collar (D0) has been installed, 'Tagziria [5]. Despite being only a few years old, the maintenance of D0 has revealed some important aspects pertinent to here mentioned problems. D0 is in continuous operation for 5 years. As part of the EURATOM safeguards scheme, three other neutron coincidence collars, one multiplicity scrap counter and one $LN_2$ cooled high purity Germanium detector are permanently installed in Melox. There is also a number of $^3$He neutron monitors and video surveillance cameras in Melox. However they are not directly related to the purpose of this paper and will not be further discussed.

#### 2.1.1. Analysis of performance of measurement stations and of data acquisition network

In general the performance of two measurement stations in Melox (D1 and D2) may be characterised as very good. We had practically no unplanned stops and losses of the data. Quality of gamma measurements was good and the electrical noise low. On contrary, the neutron detector D0 has lost some of its efficiency recently. The electrically cooled Germanium detectors belong to the first generation of their kind – cooling is basically done by cryopump. With annual maintenance of the cryopumps, ageing of the gamma detectors did not raise any specific problems except for the high and

rising costs to maintain them. However, after more than 15 years, the end of life of the cooling apparatus was close. On the other hand, the data acquisition and transmission network in Melox was relatively complex and consisted of more than 10 data acquisition cubicles, long cabling and a number of other network elements: switches, converters, etc. Because of short intervention time slots in controlled areas individual network elements were modified as further network development required or routinely replaced with available spares when they were broken. With the long time of operation, also core network components had become obsolete. Communication breakdowns became more frequent in Melox. However due to the solid redundant set up of the measurement setups and measurement data backups on local data acquisition computers no data was lost and continuity of knowledge maintained. The summary of performance of EURATOM's measurement and data acquisition network is as follows.

Positive points:

- Good performance of neutron and gamma detectors (except of D0 neutron coincidence collar) over many years, good measurement quality, low EMI noises,
- Data acquisition redundancy working successfully (primary and secondary computers),
- Backup of measurement data on servers located in the EURATOM office,
- Monitoring of health status of data acquisition computers,
- Automated data acquisition and analysis tools.

Weak points:

- Ageing of gamma detectors and increasing costs of their maintenance,
- Partial loss of efficiency of D0 neutron coincidence collar (detector is located in extremely difficult-to-access area and its maintenance is limited to a few days per year – during the plant's outage),
- Ageing and obsolescence of data acquisition computers and I/O electronics,
- Dominant star-type network where the EURATOM Office is connected by individual links to data acquisition cubicles in the plant; in case of a failure of one network segment the communication to one or several data acquisition cubicles is disabled,
- Central monitoring of all key network components was not possible due to diverse and often incompatible and obsolete devices,
- Limited data throughput due to some coaxial segments of the network.

In respect to the improvement of performance of the measurement stations and of the data acquisition network in Melox the modernisation primarily needed to be oriented to data acquisition networks. Replacement of two first generation type electro-mechanically cooled high purity Germanium detectors (D1 and D2) was desirable. Investigation of loss of efficiency of D0 detector was necessary and its efficiency needed to be restored.

## 2.2. Plutonium stores at La Hague Reprocessing Plants

The plutonium oxide product stores of two reprocessing units, UP2 and UP3, are in the scope of this article. Four unique measurement stations consisting of the neutron coincidence collars and of electro-mechanically cooled high purity Germanium detectors are installed at the entry and the exit of the plutonium store of UP3 plant (ENTRY A, ENTRY B, EXIT A, and EXIT B), 'Zukosky [6]. Two measurement stations made of custom designed neutron coincidence collars and high purity Germanium detectors are installed at the entry and exit of the plutonium store of UP2 plant (ENTRY, EXIT). Four UP3 measurement stations have been in a continuous operation for more than 20 years and two UP2 measurements stations for around 15 years. Several $^{3}$He neutron monitors and video surveillance cameras as well as electronic seals are used in UP2 and UP3. These are not covered by this paper.

### 2.2.1. Analysis of performance of measurement stations and of data acquisition network

The performance of measurement stations in La Hague was less satisfactory than in Melox. Measurement data losses especially for gamma detectors were longer and more significant. In general, the quality of the signals from gamma detectors used to be always poorer compared to the identical D1 detector of Melox (the same La Hague cans are re-measured in Melox). A separate study has been ordered by DG ENER and produced by JRC ITU Ispra in 2009. It allowed us to understand the reasons of poor performances of electro-mechanically cooled high purity Germanium detectors in

La Hague (UP2 and UP3) and to look for appropriate solutions for improvements. The study has proved that electromagnetic interference (EMI) was an essential problem related to the gamma detectors, 'Berndt [7, 8]. JRC ITU Ispra has also proposed a distinctive technical solution to minimize EMI. A Faraday cage for the new type gamma detector has been constructed and successfully tested in the laboratory and under plant conditions.

Although occasional issues with neutron coincidence collar measurements were not as important they have become more frequent recently due to ageing of electrical components of the collars.

In the cases where the gamma measurements are not available the inspectors may use the isotopic composition of plutonium declared by the operator. Samples from every batch of materials are independently analysed and verified by the destructive analysis methods at EURATOM on-site laboratory. Operation of EURATOM's on-site laboratory is in details described by 'Blohm-Hieber [9].

All key measurement points are located in a close range in UP3 and in UP2 and the number of data acquisition cabinets is not big (UP3 – 2, and UP2 – 4). This is why the data acquisition networks in UP2 and UP3 (both plants have separate networks and EURATOM offices) are very simple and well organised. Nevertheless, some obsolete components of the networks needed to be upgraded or replaced. Summarizing the overall performance of EURATOM measurement and data acquisition and transmission networks, the following can be said.

Positive points:

- Simple and well organized data acquisition and transmission network,
- Good performance of neutron coincidence collars over long years,
- Data acquisition redundancy (primary and secondary computers),
- Backup of measurement data on servers located in the EURATOM offices,
- Monitoring of health status of data acquisition computers,
- Automated data acquisition and analysis tools.

Weak points:

- Ageing of gamma detectors and increasing costs of their maintenance,
- Recent instant efficiency losses of some neutron coincidence collars due to ageing,
- Low geometrical efficiency of gamma detectors,
- Electrical noise measurements performed by JRC ITU Ispra have shown important EMI disturbances in the region of 2.5 - 4.5 MHz having an influence on increased dead-time of detector and decreased quality of the gamma spectra (**Fig. 1**),
- Long cabling between detectors and gamma spectroscopy electronics further weakens the signal's quality,
- Ageing and obsolescence of data acquisition computers and I/O electronics.

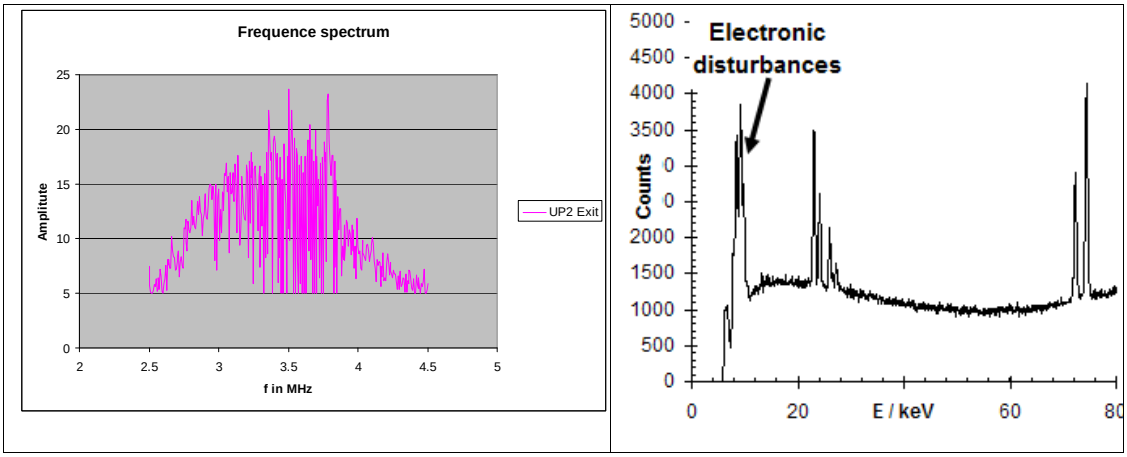


**Fig. 1:** EMI with electronics of EURATOM's gamma detector in UP2 and its influence on gamma spectrum [10].

To conclude, the performance of measurement stations, especially as concerns the gamma detectors needed to be seriously improved. The data acquisition and transmission networks could be improved in advance. However the data loss risk linked to the networks is not essential. The aforementioned study shows that the composition of plutonium isotopes due to the higher burn-ups and retreatment of spent MOX fuel will likely change in near future. The abundance of $^{239}$Pu isotope may diminish (below 50%) and the abundances of $^{238}$Pu and $^{242}$Pu isotopes increase (accordingly up to 4.5 % and 11%). The uncertainties related to the analysis of gamma spectra especially due to bigger abundance of $^{242}$Pu and lower abundance of $^{239}$Pu would become larger. This is why the statistical uncertainty related to the measurement setup (due to geometrical efficiency and dead-time of detector) required to be optimised and decreased to as low as possible. It these circumstances the analytical capabilities of EURATOM's on-site laboratory to verify the isotopic composition of plutonium will remain essential.

## 3. Improvements of EURATOM's unattended measurement stations

The aforementioned measurement stations were unique and have been developed and installed starting from 1991. At that time, this was pioneering work. The majority of them provided good performance and operation for over 20 years without major maintenance! However some of them have shown problems. Recently we have been observing some performance's deterioration of some of the stations. Analysis and understanding of the problems related to the measurement stations in Melox and in UP2 and UP3 enabled us to draw some decisions: all electro-mechanically cooled high purity Germanium detectors have to be replaced! The study performed by JRC ITU Ispra [7, 8] has shown that in this case good substitutes are the new generation Cryo-Pulse® 5 Electrically Refrigerated Cryostats equipped with high purity Germanium detectors (CP-5), 'Data Sheet [11]. Some previous years of operation of CP-5 in nuclear facilities under EURATOM control as well as the positive experience with CTBTO which operates many CP-5 systems all over the world in remote measurement mode proved them to be a very suitable solution for difficult-to-access areas where the maintenance time is very limited, 'Cella [12]. Thus, the maintenance costs are significantly reduced. For the time being 6 of 8 old generation gamma detectors have been already replaced with the CP-5.

Some decisions like the replacement of the polyethylene bodies of all four neutron coincidence collars in UP3 were rather taken by precaution. We were uncertain about the physical integrity of the polyethylene. The risk to lose the neutron coincidence collars because of polyethylene's embrittlement and impossibility to use it again was not acceptable.

Our recent experience confirmed us that the investments in the modernization of the data acquisition and transmission networks were very beneficial and gave us very good return, especially in case of Melox.

### 3.1. Melox MOX Fuel Fabrication Plant

Two old electro-mechanically cooled high purity Germanium detectors were replaced with two CP-5 detectors by June 2012. Different sub-contractors were involved in the project and the work was completed with a slight delay. Due to access limitations, especially as concerns D1, all interventions had to be planned well in advance. The replacements have also involved mechanical modification of detectors' supports and supplemental cabling. In addition the following improvements were implemented which increased the reliability of the new detectors and of the data acquisition:

- The supports were equipped with silent-blocs to minimise the impact of vibration to the detector's head,
- Isolators were used to create galvanic separation between the supports and the detectors,
- Heat evacuation of EURATOM data acquisition cubicles was improved,
- Old acquisition electronics were replaced with MCA-166 multichannel analysers 'Gabriel [13]; this modification permits also full remote control of the HV and of inhibit lines,
- Detectors were equipped with two identical preamplifier outputs (primary & secondary),
- Redundant data acquisition computers were installed to acquire independently the primary and the secondary signals of both gamma detectors.

The two neutron coincidence collars (D1 and D2) were not modified. Only modification was related to the measurement data acquisition and the replacement of computers. D0 had shown a change in

efficiency. Maintenance of the D0 neutron coincidence collar took place during the outage period of the Melox plant in January 2013. Contributions from individual tube banks (there are ten banks grouping four $^{3}$He tubes each) to the total counting of the collar have been measured and, as it appeared, they were not equal. One bank of tubes was replaced and the initial efficiency of D0 detector was restored. However we have not yet established the reason of this uneven contribution. One suspect is HV capacitors. Unfortunately, the D0 key measurement point is one of these areas where the in-situ maintenance time is very limited and we could not perform thorough tests and in depth maintenance to prove this supposition. On the other hand we do not think that the $^{3}$He tubes having extremely long MTBF (mean time before failure) may fail for relatively young detector D0. Further investigations to mitigate the issue are in preparation.

Earlier star-type EURATOM data acquisition network has been largely upgraded and modified to the ring-type network (**Fig. 2)**. The benefit of the latter is that every location in plant is connected by two independent fibre-optical connections. If for example one fibre-optical network segment cable fails there is still another connection available. More details about the advantages of this type of networks and further developments are presented by 'Schoop [14]. New centralised network watchdog have made possible automated surveillance and regulation of the data acquisition network. This network modernization has also reduced the number of network elements. Some excessive network elements were removed so making the network simpler and more resistant to single-failures. Old coaxial cables were replaced by multi fibre-optical cables. Finally, we have decided to replace all old data acquisition computers and to create more redundancy for data acquisition where possible. As a result of this modernization the network failures involving total loss of the network communication have practically disappeared.

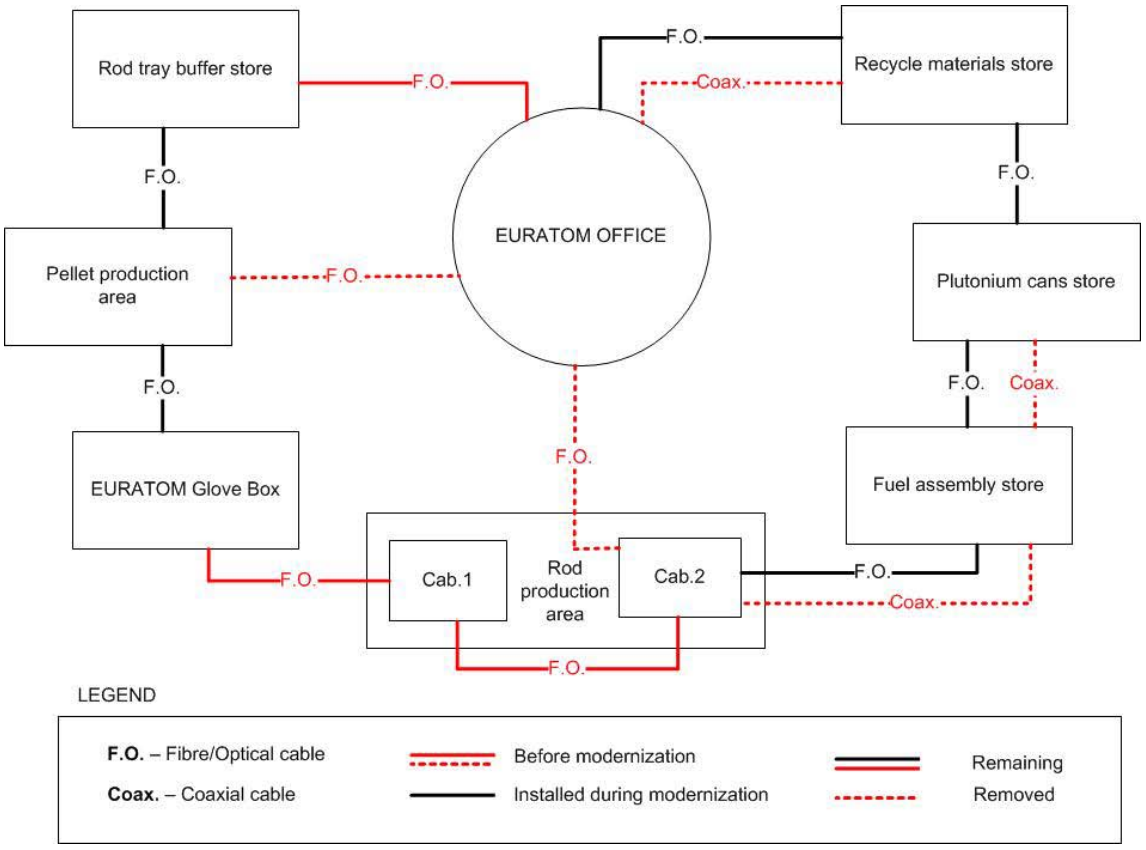


**Fig. 2.** Modernization of EURATOM's network in Melox MOX Fuel Fabrication Plant.

We are continuously improving the reliability and availability of EURATOM systems in Melox. The above-mentioned works were realized in 2012 in parallel to the normal maintenance operations. Frequent presence on-site allowed us to dedicate special attention to inventorisation and documentation of EURATOM's equipment in Melox. The prioritized lists of spare parts and the needs for preventive maintenance were established.

### 3.2. La Hague Reprocessing Plants

The design of four measurement stations in UP3 and of station D1 in Melox was identical except for the collimation of the gamma radiation. When designed, two $^{3}$He tubes were initially removed in the D1 neutron coincidence collar in Melox and the gamma detector head was fully inserted to the collar through a large hole in the wall of the polyethylene body. A small 8 mm collimator was used for collimation, and the distance between the cylinder of plutonium cans and the Germanium crystal was short (2 – 3 cm), providing a good solid angle for the gamma flux. Whereas, in case of the UP3 systems the number of $^{3}$He tubes in the collar remained unchanged (18). A narrow horizontal hole (16 mm) was drilled into the polyethylene moderator between the central space of the collar and the gamma detector which was placed on the external wall of the collar. The measurement conditions were even worse when the cylinders with maximum 5 cans inside were positioned during the measurement in such a height that only the gap between two stacked cans was in front of the collimator hole. Similar unfavourable measurement conditions occurred if a partially filled can was placed in front of the hole. Moreover, the collimator geometry at UP3 caused rather low count rates. Our evaluations have shown that the count rate should be some thousand cps higher to optimise the measurement setup. Both problems were solved by the means of an increased collimation opening. The new polyethylene bodies have vertical trapezoidal-shape slits between two neutron counter tubes which allow for the detector's crystal to measure gamma radiation from a 30 to 50 cm high section of the cylinder. Since the gamma measurements are done in three heights it is expected that a gamma spectrum will be measured even in the case where only one of the internal cans contains $PuO_2$ powder.

The circumstances which led us to the decision on replacement of all polyethylene bodies were related to our former experience and also to the strict schedule of upgrade and maintenance works. First of all, we have already been faced to the embrittlement, deformations and fissures of the polyethylene parts of neutron measurement systems. At UP3 we had no evidence that the polyethylene bodies of the collars were intact and good enough for a further application period of many years. We were neither sure that they would keep their integrity during machining for the enlarged collimation openings. Secondly, we needed urgently this upgrade of the gamma systems and limited intervention time slots did not allocate us enough time for in-depth studies.

In the end of 2011 the first measurement station has been removed (ENTRY A) and replaced by a neutron coincidence collar similar to a JCC-71 type and a new set-up of the gamma spectrometer. It consists of a planar gamma detector with CP-5 cooling. The detector, its cooling system and a MCA-166 are placed in a Faraday cage. The 230V power line is equipped with low pass filters and the serial connection to the next PC goes over an optical RS232 - RS232 galvanic separator. Like this the complete analog electronics part of the spectrometer is protected both against EMI in the form of electromagnetic waves in the environments and in the form of disturbances from the 230V net or the serial data transfer cable. A six month test period has demonstrated the stability of the first new gamma system 'Berndt [15].

The removed neutron coincidence collar has been examined. The dimensions of the polyethylene body were verified. Electronic circuits of the collar were tested and old deficient components were replaced. We have also put the order for Canberra for four new polyethylene bodies with specified collimation openings. This order has been fulfilled in the mid-2012. First full replacement of the whole measurement station (EXIT A) took place in August 2012. Then the old removed collar (from EXIT A) has undergone the examinations and replacements of the polyethylene body and of old electronic components. After the maintenance the collar and the new CP-5 gamma detector have been installed at EXIT B in the end of 2012. We have followed the same way with the replacements of ENTRY B and ENTRY A and in March 2013 the modernization of all four measurement stations were successfully completed. All four new measurement stations demonstrated excellent measurement results and fully met our expectations. This modernisation permitted us not only to improve the reliability and availability of the measurement stations but also to minimise their maintenance costs.

Important innovations implemented during this modernisation are listed here:

- Improved collimation of gamma measurements by the means of vertical trapezoidal slits enlarging the solid angle of the measurement and the usage of adjustable lead collimators,
- New CP-5 detectors, which replaced the old electro-mechanically cooled high purity Germanium detectors were equipped with the Faraday cages which reduced the EMI and its contribution to the dead-time; it has also improved the quality of the gamma spectra,
- Other EMI reduction means were also implemented: shortening of detector's signal cables and placement of gamma acquisition electronics as close as possible to the detector, usage of galvanic isolators and differential signal converters (RS232/RS422) for the communication lines (~30 meters) with the acquisition computers, usage of galvanic isolators for the Faraday cages,
- All new CP-5 detectors have 200 $mm^2$ and 10 mm thick planar Germanium crystals, which improved the resolution of the gamma spectra,
- The new collimator design guarantees that a gamma spectrum is measured in all cases, even if there is only one internal container filled with $PuO_2$ powder,
- An adjustable lead collimator allows adjusting easily the efficiency of the set-up to the needs of the routine measurements,
- Old gamma acquisition electronics were replaced with MCA-166 multichannel analysers; this has also enabled full remote control of the HV and of inhibit lines.
- All electronic components of four neutron coincidence collars were thoroughly investigated and maintained; the laboratory measurements have revealed that the high parasite current of HV capacitors caused some fake counts by coincidence electronics; all HV capacitors were replaced.

Modernization of EURATOM measurement networks still continues. Preparatory work for the replacement of other two measurement stations in UP2 is going on. Both stations are located in the difficult-to-access areas where the intervention time slots are short. Design and types of the measurement stations in UP2 are different from ones in UP3 and the replacement works need to be well planned. The aim is to apply the same technical solution, i.e. the same construction of the Faraday cage, for the gamma spectrometer. Like this it is possible in case of maintenance to change a complete Faraday cage in one single intervention, independent on the place in UP2 or UP3.

As concerns the data acquisition and transmission networks in UP3 and UP2 some further improvements are in our focus. First of all we are planning the replacements of the acquisition computers and later some minor improvements of the cabling (UP2). However these small modifications will be implemented within the routine maintenance visits and will not require important additional resources. The inventory list and documentation is being updated.

## 4. Conclusions

After more than 15 (Melox) and respectively 22 (UP3) years of partially very successful operation, extensive modernisation of six EURATOM measurement stations and data acquisition networks has been carried out in Melox MOX Fuel Fabrication Plant and in UP3 Reprocessing Plant. All six first generation electro-mechanically cooled high purity Germanium detectors were replaced with the new type modern CP-5 gamma detectors. It has largely improved their performance and reduced the maintenance costs. A number of modifications aiming at optimization of the gamma measurement setups and improvement of the quality of gamma spectra were realised, especially in case of UP3. New successful technical solutions lead to significant EMI reduction. The radical solution with the application of a Faraday cage may be a good concept also for other measurement positions in industrial environments, where the amount of electronic components with emission of high frequency disturbances will further increase in future. Some neutron coincidence collars were exhaustively maintained and their ageing components like the moderating polyethylene bodies or aging HV capacitors were replaced. Important improvements related to the reliability of the data acquisition and transmission networks were implemented. Obsolete network components were replaced and the structure of the network modified in Melox aiming at improved continuous availability and redundancy of the data acquisition and transmission.

## 5. Acknowledgements

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## 7. Legal matters

# Towards effective safeguards implementation in the geological final disposal of spent nuclear fuel

**M. Murtezi, H. Dratschmidt, W. Kahnmeyer,**
**C. Koutsoyannopoulos, P. Schwalbach, A. Smejkal, M. Thomas, A. Zein**
European Commission – Directorate-General for Energy
Direction E – Nuclear Safeguards, L-2530 Luxembourg
E-mail: Mentor.Murtezi@ec.europa.eu

***Abstract:***

*Development of the geological disposal systems in some European countries is entering the implementation phase. This calls for a parallel development of adequate safeguards concepts and applicable techniques by the relevant regulatory bodies and safeguards authorities. The purpose made safeguards concept developed by the European Commission' Safeguards Services is primarily aiming at safeguarding the process leading to isolation of the spent nuclear fuel in a suitable geological environment. The continuous nature of this process, in combination with the fact that the nuclear material, once encapsulated and disposed, will not be available for re-verification, imposes a series of stringent requirements on the safeguards measures to be applied. Precise, efficient and reliable verification of the spent fuel to be disposed, as well as unconditional retention of the continuity of knowledge between its encapsulation and deposition, are seen as crucial.*

*Multi-party cooperation is indispensable in this challenging task and the inspectorates of the international safeguards authorities (Euratom, IAEA) have liaised from the inception stages with the national safeguards authorities and the operators of the new installations in Finland and Sweden. Safeguards-by-Design principles are being employed in a jointly developed process, aiming to assure the safeguardability of the novel installations.*

*Safeguards-by-Design addresses the complexity of the facilities and is expected to allow for an effective implementation making use of synergy effects between the needs of operators, state regulators, EURATOM and the IAEA at the earliest stage possible. Practically we apply a consultative process of design shaping, through which a geological disposal system becomes inherently safeguards-ready.*
*In the working safeguards model of the European Commission, the underground disposal facility together with the enveloping rock masses that will confine and contain the deposited spent nuclear fuel, are treated as "black box". The design characteristics and the applied safeguards measures must ensure that the underground storage environment remains untouched. All penetrations leading to it must be kept under safeguards control.*
*The paper presents the major components of the proposed safeguards system and the participatory process of its development and implementation, as well as its interplay with safety and security aspects of the geological deposition and isolation of the spent nuclear fuel.*

**Keywords:** final disposal, spent fuel, measurement systems, geological repository

## 1. Introduction

The concept of the geological final disposal (GFD) of spent nuclear fuel (SNF) has been developed on the basis of taking on the appropriate responsibility as well as responding to the social concerns on the long term safety of the nuclear waste storage. The Directive 2011/70/EURATOM of 19 July 2011 establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste addresses this concern and obliges the EU Member States to apply adequate systems for long term nuclear waste storage. In principle, the finally disposed SNF should ultimately be irrelevant to the biological environment and thus, would ultimately no longer be under the safeguards umbrella. Only the confining structures and systems including the man-made barriers and

in case of geological disposal, the enveloping geological environment, would remain under supervision. Such supervision must be of the three-dimensional safeguards-safety-security nature.

Finland and Sweden are the first two EU countries to dispose finally of their spent nuclear fuel in an underground repository. These two countries will actually be the first in the world to fully implement the GFD concepts. During the last three decades the Swedish Nuclear Fuel and Waste Management Company, SKB, developed a multi-barrier approach for the isolation of the SNF. The SKB concept based on encapsulation of the SNF into a copper canister, then buffering the canister by a layer of bentonite clay and placing it into a crystalline rock environment will be used in Sweden as well as in Finland. Following more than 25 years of site investigations, the Finnish parliament ratified in 2001 the decision in principle to construct the GFD in Olkiluoto. Excavation of the rock characterisation facility ONKALO started in Olkiluoto in 2004 by the operator company Posiva Oy. In 2009 Sweden took a decision to localize its GFD site at Forsmark. According to the project delivery plans the Finnish GFD facility should be put in operation in 2020 and the Swedish GFD in 2027.

Apart from the two advanced projects in Finland and Sweden, countries like Belgium, France, Germany, Switzerland and the UK are also developing their own concepts for GFD and carry out site investigations.

The GFD concept is now in the transitional phase from its development to implementation. This is the crucial moment for planning of the applicable safeguards concepts that shall meet the requirements laid down by the national and international law, being at the same time effective and efficient in front of this long lasting and multi-staged project. The ongoing process of shaping the safeguards approaches is progressing in a participatory and consultative way involving the following major actors:

- The inspectorates of the international nuclear safeguards authorities: the European Commission and the International Atomic Energy Agency
- The nuclear power plants' operators from Finland and Sweden
- The companies responsible for the construction and operation of the geological repositories
- The national authorities involved in safeguards
- The international research community

This multi-party consultative process is in an advanced stage and the general safeguards concepts have already crystallised, opening discussions on the necessary technical arrangements.

This paper presents the concept and applicable safeguards implementation techniques, at the current state-of-the-art. The entire concept must be adaptable to the technological progress in the future and is thought to be adaptable and scalable to specific GFD projects and different stages of the project's life cycle.

## 2. Geological disposal - a challenge to safeguards

The GFD process can be characterised by the following safeguards-relevant intrinsic features:

- its novelty; Finland and Sweden to be the first in the world to construct and operate GFD
- timescale; tens of years of operation and the geological timescale to be applied when speaking about the retention time
- multistage implementation; construction phase, operational phase and the post-closure stage
- high throughput; during the operational phase (reaching 200 canisters disposed per year)
- large capacity; thousands of canisters to be deposited
- practically no re-verification possibility under current conditions; in the geological stage, a potential access to the deposited nuclear material would have become very difficult and extremely costly.
- natural barriers playing a major role in isolation and access restriction; the SNF will be deposited at depths of more than 400 metres in a crystalline rock environment.

The above characteristics make the GFD a technologically complex and a very expensive project. Finland and Sweden took on a great challenge to be pioneer constructors in this novel area with its technological, legal and managerial complexities. The same types of challenges emerge on the side of the safeguards authorities' inspectorates. The concept of the GFD, as a state-of-the-art way of assuring maximum long term safety and security of the SNF retention, requires a robust safeguards approach able to meet requirements at the different stages of the GFD implementation. Meeting all the provisions of the safeguards-related legal acts and using the currently available techniques, the applicable safeguards system must be of an open and adaptable nature. This is dictated by the timescale of the GFD projects stretching over several generations.

The GFD projects in Finland and Sweden are implemented with "safeguards in mind". This safeguards-by-design (SBD) approach vastly facilitates application of safeguards and integration of its systems with the physical system components but also with the safety and security arrangements.

At present the EU safeguards inspectorate puts a lot of effort in assuring a tight grip on the nuclear material within the nuclear fuel cycle: from mine to conversion, enrichment, fuel fabrication, reactor to storage pond and further to the interim store and reprocessing plant where appropriate. This final disposal concept adds additional stages to this cycle: encapsulation of spent fuel and its transfer to the underground location and emplacement in its defined position.

The overall goal of the EU's nuclear safeguards system is stated in chapter VII of the Euratom Treaty and the Commission Regulation (Euratom) No 302/2005. The major goal of safeguards measures is to know and verify who, where, what and how much of nuclear material is used for what purpose and to verify, if the reality in the field or on-site is correctly reflected in the operators' declarations. In practice, all nuclear material in Europe is subject to Euratom safeguards and must be strictly accounted for. The accountancy declarations must be verified via the on-site inspection activities, aided by remote monitoring techniques. In front of the GFD these questions can be satisfactorily answered at all times under the conditions that all the material that goes underground is correctly accounted for and is strictly retained therein after it enters underground.

The principal question that must be answered before planning a safeguards system for the GFD is: what is the goal we want to achieve? In other words: is it enough to know how much nuclear material is entering the underground to be deposited or does the GFD introduce any new safeguards needs, such as the necessity of constant monitoring of the underground environment after the SNF is deposited – and if yes, to what extend? The safeguard system must be relevant in front of this goal and the effective particular safeguards provisions must be established and adequate and efficient technical solutions adopted. Last but not least, the capacity of the safeguards inspectorates must be taken into account while choosing the safeguards solutions. The GFD process adds an additional stage to the nuclear fuel cycle and calls for an additional capacity of the system responsible for its safeguarding. A high level of automation of the data acquisition and analysis can, to a large extent, facilitate this labour-intensive process, nonetheless, especially during the construction and in the beginning of the operational phase, a lot of effort will be needed from the national and international safeguards authorities.

The final test of all of the GFD and safeguards systems components will only be possible with the beginning of the operational phase. The perspective of the post-closure phase is so distant, that at this stage it is difficult to envisage any particular safeguards arrangements without entering into a philosophical discussion. However, the safeguards-by-design concept tries to address the post-closure state of the GFD facilities; their post-closure safeguardability should be assured by their intrinsic design feature. The timescale of the GFD process and non-excludable possibility of retrieval of the previously deposited SNF requires adequate techniques for attributing very long lasting signatures to the canisters to be disposed of. Another challenging task is to ensure activation of the data necessary for recognition of any potentially retrieved SNF over a very long period of time. The signatures and the database should be readable at least until the end of the operational phase if not beyond. Any additional requirements and safeguards choices to be applied during the post-closure stage will have to be designed in the perspective of a few tens of years.

## 3. The model system for the geological final disposal

Based on the systems developed in Finland and Sweden, the GFD is realised in two major steps:
1 – encapsulation of the SNF carried out in the spent fuel encapsulation plant
2 – deposition of the encapsulated fuel in the geological repository

## 3.1. Spent fuel encapsulation plant

The spent fuel encapsulation plant (SFEP) is an interface installation between spent fuel storage facilities or reactors from which it receives shipments of spent fuel in the form of spent fuel assemblies and a geological repository (GR) to which it ships filled disposal canisters. Its major purpose is the conditioning and packaging of spent nuclear fuel into a form suitable for final disposal.

The model SFEP consists of the following main modules:

- receiving area;
- hot cell area with:
    - fuel handling cell
    - measurement and drying stations, possibly with buffer capacity
    - canister welding, filling with noble gases, welding machining, decontamination, and inspection stations;
- shipment area.

Apart from these main modules, a SFEP includes also a store with empty disposal canisters, a buffer store for filled disposal canisters, staff facilities and the main control room.

Spent fuel elements arriving directly from a reactor pond or from an interim store are delivered to the receiving area in a transport container. In the fuel handling cell fuel assemblies are retracted from the transport container, disassembled (could be the case in some of the SFEPs), measured and dried of any residual moisture. At this stage the fuel assemblies can be verified for the last time without major restrictions before being placed in a disposal canister.

The inspected and verified fuel assemblies are filled into a disposal canister. When fully filled the canister is permanently closed using inseparable connection techniques, such as spin welding or electron beam welding. The welded connection is later inspected and the canister is transferred to the shipment area of the encapsulation plant. At this point a radiometric fingerprint of the canister and/or of the weld seam can possibly be taken for later identification.

From the shipment area the canisters are transferred to the GR either by lift (Finland), in case where the repository is adjacent to the SFEP, or by means of reusable shielded transport flasks when the GR is located away from the SFEP (Sweden).

The integrity of SNF brought to a SFEP should be unchanged, i.e. no manipulations of the fuel assemblies other than reloading it from transport containers into the disposal canisters are to be carried out at SFEPs, with the exception of the aforementioned disassembling if applied. Under normal operating conditions the SFEP has unidirectional flow of nuclear material although shipment of the rejected fuel assemblies may occur.

## 3.2. Geological repository

A geological repository (GR) is an underground facility designed for the final disposal of the already encapsulated SNF. The GR consists of the underground infrastructure (tunnels, shafts, transport and ventilation systems etc.) as well as the enveloping geological rock environment. The enveloping rock masses confine and contain the deposited spent fuel in its encapsulation and should be treated as integral part of the GR.

The design characteristic of a GR must insure total isolation of its underground environment and maximum control of all the penetrations leading to the underground storage. This control must be applied at the surface as well as to the rock volume surrounding the repository.

The term "final disposal" should be understood as a way of confinement capable of isolating SNF from the biological environment for a very long period of time (hundreds of thousands of years). In principle the geologically confined, i.e. finally disposed SNF is in this current concept treated as irretrievable. This may change with time and politics of the future, nonetheless, the final verification of the SNF to be disposed of should meet the most stringent requirement which is assuming irretrievability. On the other hand, as discussed above, the retrieval possibility calls for reliable and very long-lasting signatures of the deposited canisters as well as for a robust way of data archiving and long term readability (recognition and compatibility with future soft- and hardware).

The model GR consists of the following main components:

- surface-located disposal canisters reception area with the canister transport shaft inlet
- system of tunnels and shafts for personnel and material transportation
- ventilation and personnel emergency evacuation shafts
- buffer store for the received disposal canisters
- system of disposal corridors where the canisters are emplaced in drilled deposition holes
- underground laboratory
- material and equipment storages

Apart from these components a GR comprises also the geological environment in which it is located.
A GR must be located within a stable lithotectonic unit consisting of rock formations providing adequate isolation from chemical and mechanical factors as well as from any unauthorized access. The purpose of this isolation is to insure integrity of the deposited canisters over a very long time and provide a robust barrier between spent nuclear fuel and the biosphere. In the post-operational phase the SFEP and ultimately all the ground-based structures would be dismantled. After backfilling of the underground tunnels and shaft the GR would be restricted to the volume of the confining rocks.

## 4. Safeguards concept

Firstly, the adopted safeguards system for the SFEP and the GR should be technically feasible and appropriately applicable. This statement may seem obvious and redundant, nonetheless, without the anticipatory consideration of all the boarder conditions, such as throughput of the installations, buffer and anomaly-solving capacity as well as the necessary human and financial resources, the safeguards choices could be challenged and made obsolete by the reality. Secondly the system should be acceptable to all the involved parties.

The safeguards aspects of the GFD process have been discussed over many years among the nuclear safeguards community. The generic guidelines for safeguarding geological repositories were proposed in 1997 based on work of the IAEA Working Group for the Development of Safeguards for the Final Disposal of Spent Fuel in Geological Repositories (SAGOR) [1]. The work of the SAGOR-I (1994-1998) and SAGOR-II (1998-2005) is continued since 2005 by the expert group on the Application of Safeguards to Repositories (ASTOR).

In 2010, in order to assure the feasibility, the applicability and the acceptance of the proposed solutions, a forum for information exchange concerning GFD projects in Finland and Sweden was formed grouping the six involved parties:

- EC DG ENER-E
- IAEA
- STUK (Finland)
- Posiva Oy (Finland)
- SSM (Sweden)
- SKB (Sweden)

The applied safeguards systems should ensure that the SNF to be finally disposed of is verifiable and precisely accounted for up to the moment when it becomes encapsulated. From the moment of transfer of the disposal canister to the GR the safety and security systems must ensure that the SNF remains inside an intact multi-barrier based confinement. i.e. adequately confined and isolated from the biosphere. Aiming at the assurance that all the nuclear material declared to be geologically

disposed of is appropriately safeguarded at all stages until it goes underground where it stays, the safeguards system, while meeting its own goals, can complement and contribute to the safety and security efforts. To achieve these goals, the development of the safeguards approaches and techniques had started in parallel to the development of the final disposal concepts and technologies. Application of the safeguards-by-design principle allowed for bi-directional feedback between the designers and the operators of the GFD installations on one side and the safeguards inspectorates on the other.

The SBD is aiming at the flawless effective and efficient verification of the SNF to be encapsulated and deposited and the 100% assurance of continuity of knowledge (CoK) between the encapsulation and the deposition of the SNF. The safeguards system and measures should have the least possible impact on the envisaged capacity and throughput of the GFD process while allowing each safeguards authority to effectively fulfil their mandate imposed by European, international and/or national law.

A key point in the elaboration of the safeguards approach for final disposal was the definition of the location in the process where the final verification measurement of SNF can be made. The two following options have been examined:

- At the SNF shipping facility = the NPP spent fuel storage
- At the SNF receiving facility = the SFEP

Taking into account the legal requirements and objectives of the safeguards systems to preserve the CoK with the help of containment and surveillance measures until the disposal canisters are placed underground, the safeguards inspectorates (the EC and the IAEA) have opted for the final verification of the SNF to be done at the encapsulation plant. As a result, the SFEP is the focal point of the proposed safeguards system. After the final verification and encapsulation of the SNF until its entry into the underground environment of the GR, it will remain under constant radiation monitoring and optical surveillance. Within the EU this concept was first developed for the German Pilot Conditioning Facility PKA at Gorleben in the late 1980s [2], [3].

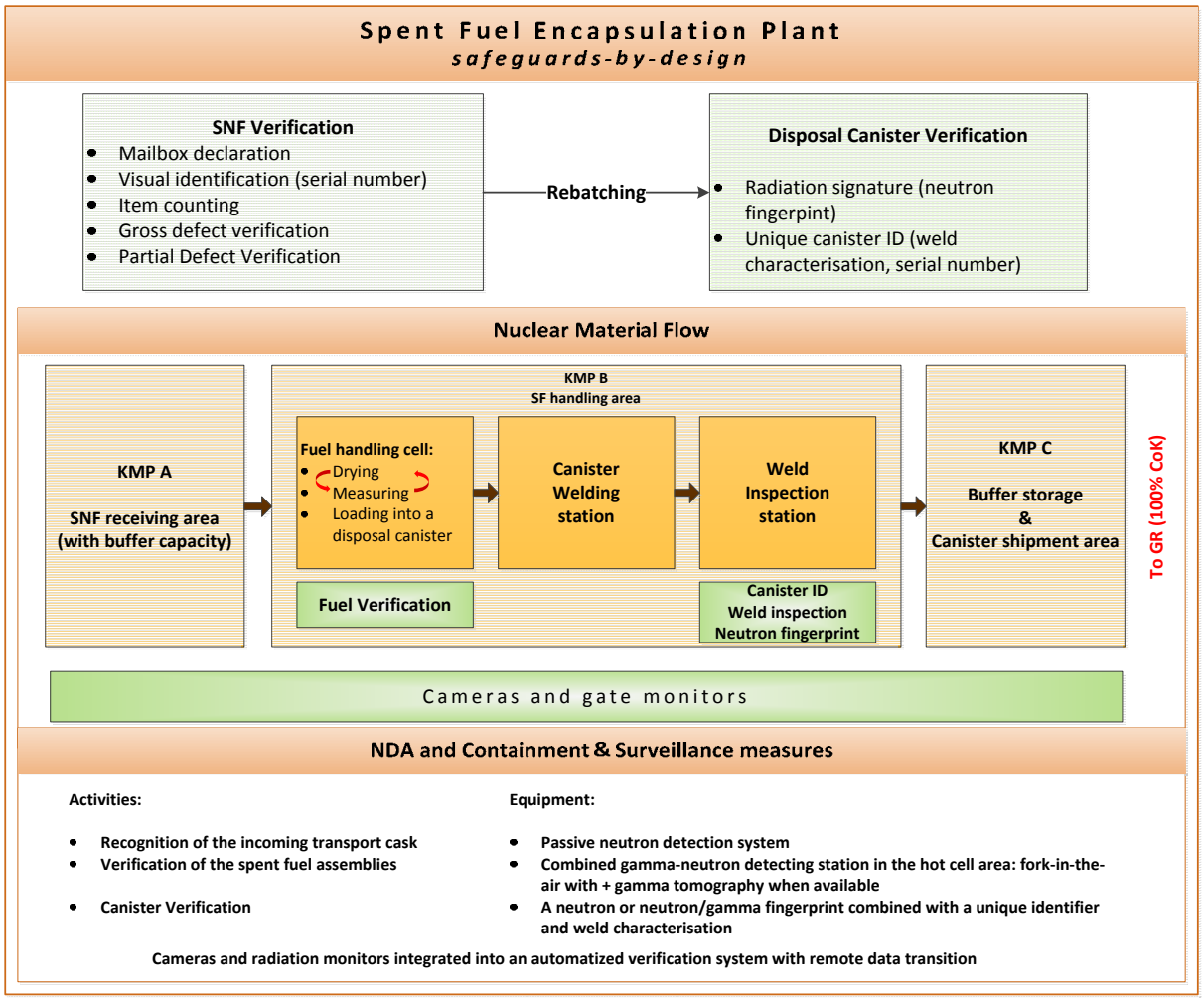


Figure 1: Stages of the GFD process in a model SFEP and a general overview of the safeguards concept and it implementation methods.

In the proposed safeguards concept the GR is seen as a "black box". This means that all the nuclear material deposited underground must be accounted for and characterised enabling its future re-verification in the event of retrieval at least until the back-filling stage. However, the black-box GR as a whole will neither be monitored internally nor externally for safeguards purposes (e.g. using seismic, acoustic or thermal sensing techniques). No verification of the retained disposal canisters is envisaged. Portal radiation monitoring will be applied to all penetrations leading to the underground and periodical reviews of the technical characteristics of the GR will be conducted. This black-box option is in agreement with the rationale of the GR concept and its wider context of safeguards, safety and security; monitoring of the finally disposed SNF would contradict the underlying idea of its total isolation from biosphere and the self-protecting role of the GR.

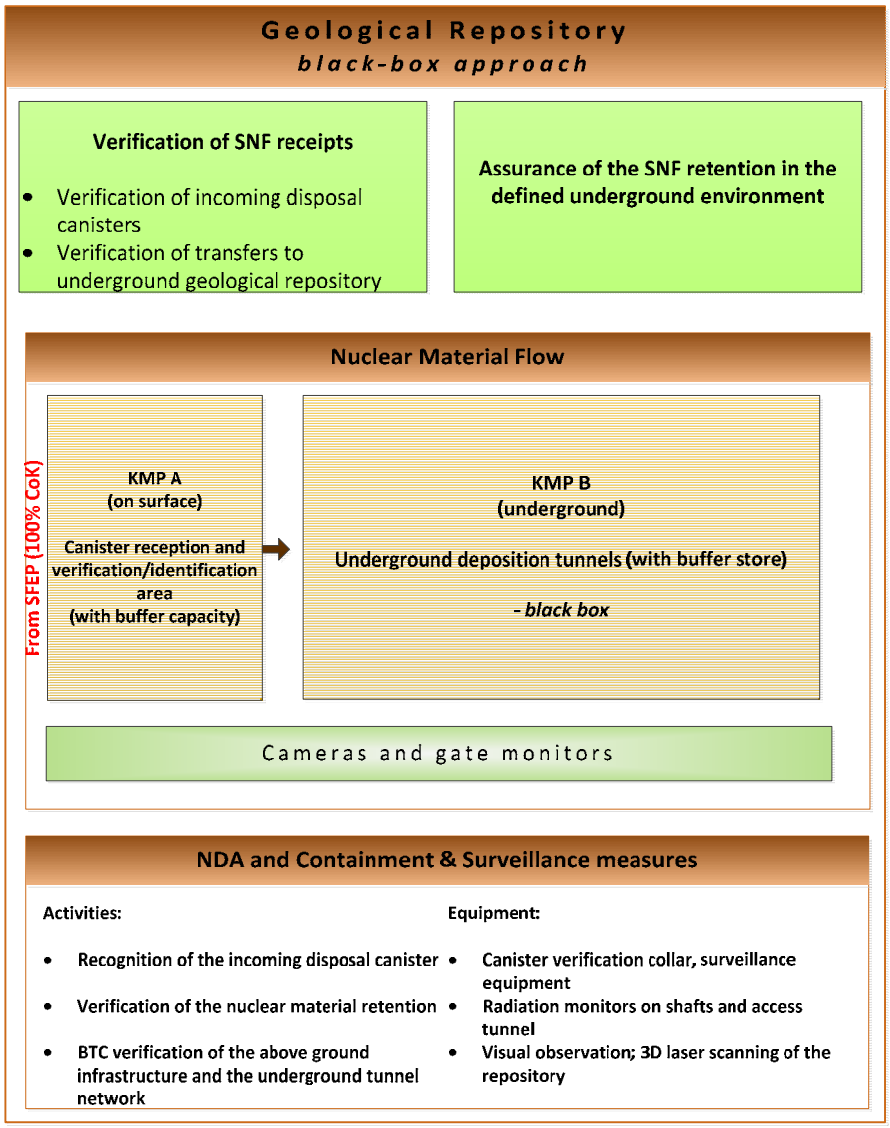


Figure 2: Stages of the GFD process in a model GR and a general overview of the safeguards concept and it implementation methods.

The seismic-acoustic monitoring as well as satellite imagery and ground water analyses could be used to assure safety and security of the GR and its surrounding. Nuclear safeguards may obviously benefit from the results of such investigations, however, the proposed safeguards system does not pose the requirement of using them. Certainly, during the operational phase of the GR there will be possibilities to verify positions occupied by the disposed canisters – this can be done as a part of the basic technical characteristics verification. At the current stage the EC is not envisaging any additional physical controls to be applied underground. The SFEP and the GR will constitute separate material balance areas for the nuclear material accountancy.

The safeguards concept being developed is addressing the operational stage of the GFD process in great detail. The post-closure safeguards measures can only be proposed in detail when the number of unknown parameters (such as the future technological developments, political decisions and social and economic constraints) will be better assessed and start to shrink and fade away with time. This will only happen at a several decades horizon.

### 4.1. The final verification and approval for the encapsulation – the crucial safeguards step

The EC safeguards effort is to focus on the state-of-the-art verification of nuclear fuel, as close as possible to the moment of its encapsulation and final disposal. The continuous and long lasting operational regime of the SFEPs stipulates a potentially labour-intensive and costly challenge for the safeguards services. The effective and efficient application of safeguards in SFEPs requires extensive use of the unattended data acquisition techniques combined with the remote data transmission. As the operational lifetime of SFEPs is in the order of several decades, the applied systems, including the storage and availability of acquired data and information, must be highly reliable and easily upgradable. The currently under development safeguards systems are tailored to the design of the particular SFEP facilities, nonetheless they have many similar components. In each case the non-destructive analysis stations are based on the best available detection method for partial defect verifications such as an Enhanced Fork Detector Irradiated Fuel Measuring System (E-FDET).

The measurement station must meet the following criteria:

- the system should be able to accept and measure all types of fuel assemblies to be processed in the SFEP
- provide enough space for adding in future additional equipment, such as the unattended gamma emission tomography (UGET), which is under design investigation;
- the equipment should be able to withstand radiation levels reaching 100 Sv/hour;
- it should be easy to decontaminate - especially important in case of using wet transport containers between the interim store and the SFEP and in case where the final verification measurement would be performed before drying;
- The on-site maintenance, i.e. replacing electronics and detectors must be taken into account when designing the measurements systems in order to reduce radiation exposure to maintenance staff.

Such a system must operate automatically and must be highly reliable in unattended mode (as far as inspectors are concerned). The system should have a constant indication of its health status available to the operator. The acquired data (neutron and gamma count rates plus the obtained spectra) would have to be transferred to the inspectorates for comparison with expected values and reference data. If the results would remain within the agreed tolerance the assemblies would be released for disposal. In order to achieve the adequate level of automation, under the action sheet 29 of the agreement signed between the United States Department of Energy and the EURATOM [4], the depletion and cooling modelling code ORIGEN by Oak-Ridge National Laboratory has been recently integrated in the EURATOM unattended measurement acquisition and review code RADAR-CRISP. A new package CRISP-ORIGEN has been created for the integration of the operator declared irradiation history in the evaluation of the E-FDET data measurements. The continuous operation mode and high throughput put tight time constraints on the measurement and validation time available to the inspectorates. The almost real-time transition of data (secured and authenticated) will be crucial to reconcile the strict safeguards requirement and the efficiency of the plant's operation. The EC in coordination with the IAEA and the state authorities and operators in Finland and Sweden are currently working on adequate technical solutions to meet all these requirements.

After the verification, the SNF is leaving the SFEP in permanently closed disposal canisters. The canisters are made of copper and can contain different numbers of fuel assemblies – depending on the SNF type. This implies the need for re-batching the nuclear material for the respective canister. The spent fuel loaded into the disposal canisters de facto becomes inaccessible (possibly a new material description code could be proposed referring to the geologically confined spent fuel) despite the fact that the operating licence for the receiving geological repositories could envisage the possibility of retrieval of the confined fuel. The radiometric fingerprint can serve as a unique identifier for each disposal canister leaving a SFEP.

The filled disposal canister may be placed in a shielded transport container and shipped by road and/or maritime transport as appropriate or may be directly transferred to an adjacent geological repository via a canister transfer tunnel. Particular arrangements for preservation of the continuity of knowledge during transport of the disposal canister to the receiving MBA must be adapted to the specific transport arrangements.

Various scenarios and technical solutions including these for solving the anomalous situations are under consideration taking into account the already available design information of the plants to be

built. The detailed technical arrangements for the Finnish SFEP must be agreed latest by 2015 when the planned construction starts. In Sweden, this is scheduled for 2017.

## 5. Final remarks

The GFD projects in Finland and Sweden are currently reaching the end of the design stage and are in the transition to the implementation phase. The approaching launch of the construction of the SFEPs requires intensified work of the safeguards inspectorates to meet the deadlines for adoption of the necessary physical components of the agreed safeguards systems (non-destructive analysis stations, material flow monitors and containment and surveillance equipment). The speed and the length of the GFD process calls for a highly automated and robust verification system able to handle anomalies effectively without slowing down the operational process and the overall throughput.

The biggest challenge in shaping the adequate safeguards concept is the reconciliation of the need to achieve the highest possible partial defect detection probability and the CoK ensurance (effectiveness) with the capacity of the safeguards inspectorates and the envisaged throughput of the GFD systems (efficiency). In other words, the proposed systems must not compromise the safeguards assurance and be applicable and feasible with the available resources.

The safeguards system being developed is based on the state-of-the-art available NDA techniques and relies heavily on the remote monitoring and data transmission. Without the use of these technologies it would be impossible to safeguard the GFD process, but even using them will be a considerable burden for the national and international safeguards inspectorates. The common use of the systems and data sharing should aid in achieving effective yet efficient safeguards implementation. It is clear that the introduction to the nuclear fuel cycle of the additional stage to be safeguarded calls for appropriate capacity increase.

Another aspect of the safeguard shaping process is the overall role of safeguards and their link to the safety and security assurance systems. This question becomes exceptionally valid in relation to the GFD that is in its current concept leading de facto to the irrelevance of nuclear material formerly under classical safeguards. At the current stage the safeguards-shaping efforts are focussed on technical solutions applicable along the path leading to the GFD. Nonetheless, there is a need for parallel development of the stronger links to safety and security aspects of the isolation of the SNF.

In conclusion:

- The geological final disposal is currently seen as the best possible way of addressing the long term safety and security of confinement of the spent nuclear fuel.

- The safeguards concepts and systems to be applied are shaped in a participatory multi-party way aiming at balanced effective and efficient safeguards solutions.

- The adequate and sustainable safeguards assurance will be achieved via the safeguards-by-design principle and use of modern safeguards techniques such as unattended measurement stations.

- The focal point of the GFD process is the final verification of the SNF that should be done at the last stage of the SFN handling, i.e. just before encapsulation. From this moment the non-compromised continuity of knowledge must be assured and documented until the moment when the encapsulated fuel is deposited in the GR treated as a "black box".

- Development of the new techniques applicable along the GFD process will continue over the next years until the cold tests in the field and beyond. The first years of operation of the GFD systems will be the time of the most intensive safeguards effort.

- From the safeguards perspective, the final disposal is seen as an additional stage in the nuclear fuel cycle and a new challenge. It will be a continuous and high throughput process

during which the nuclear material will disappear from the directly accessible verifiable locations. This nature of the GFD puts stringent conditions of capacity, robustness and scalability on the safeguards systems. In spite of the application of the technologically advanced automated safeguards, the nuclear safeguards inspectorates must have adequate capacities, i.e. adequate financial and human resources, adequate knowledge and state-of-the-art know-how in order to effectively fulfil the legal and social obligations.

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Paper is not due until Feb 2013

Abstract:

## FLASH Portals Project: Preliminary Results and Potential Impact

The "FLASH Portals Program" is a collaboration between Arktis Radiation Detectors Ltd (CH), the Atomic Weapons Establishment (UK), and the Joint Research Centre (European Commission), primarily funded by TSWG/CTTSO (US). The program's goal is to develop a technology to detect shielded Special Nuclear Materials (SNM) more efficiently and less ambiguously by exploiting time correlation. Unpublished experimental results will be presented, indicating a two fold increase in detection performance compared to neutron counting. Furthermore, the results suggest these signatures to be less susceptible to shielding than both neutron counting and gamma spectrometry. Although the results are preliminary and are being verified in Phase II tests scheduled for January and February 2013, the potential impact is significant: This method holds the potential to obtain highly selective, low background detection results from cost-effective, scalable large area detectors such as plastic scintillators. The technique has the potential to enable substantial advances in the fields of standoff detection, shielded source detection, and source characterization.

# Uranium Isotopic Measurements using a New Uranium Hexafluoride ($UF_6$) Gas Source Mass Spectrometer for Certification of Reference Materials and Nuclear Safeguards Measurements at JRC-IRMM

**Stephan Richter, Jan Truyens, Heinz Kühn, Sébastien Mialle, Yetunde Aregbe**

European Commission, Joint Research Centre, Institute for Reference Materials and Measurements, Geel, Belgium
E-mail: stephan.richter@ec.europa.eu

***Abstract:***

*The European Commission's Institute for Reference Materials and Measurements (JRC-IRMM) provides policy support in nuclear safeguards, nuclear security, nuclear safety and standardization. This support includes specific tailor-made reference materials, measurement services as well as the development of new reference measurement methods and their implementation.*
*The implementation of uranium hexafluoride gas source mass spectrometry ($UF_6$ GSMS) at JRC-IRMM started more than 40 years ago. It was based on two foundations: firstly the installation of the MAT511 GSMS instrument and secondly the preparation of a set of primary $UF_6$ reference materials, which were converted from gravimetrically prepared mixtures of highly enriched oxides of $^{235}U$ and $^{238}U$. Recently a new gas source mass spectrometer for uranium isotopic measurements using $UF_6$ gas, the "URANUS" from Thermo Fisher Scientific, was installed at JRC-IRMM, which also allows measurements of the so-called "minor" isotope ratios $n(^{234}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$. The "single" and the "double standard" method (DS) as well as the newly developed "memory corrected double standard" method (MCDS), which required a detailed investigation of memory effects within the GSMS instrument are described. The measurement performance for the "major" isotope ratio $n(^{235}U)/n(^{238}U)$ as well as the minor isotope ratios $n(^{234}U)/n(^{238}U)$ and $n(^{236}U)/n(^{238}U)$ is presented and compared with other mass spectrometric techniques. The plan for a new certification of the basic $UF_6$ reference materials such as the IRMM-019-029 series in particular for the minor isotope ratios is outlined. With the installation and the validation of the new URANUS $UF_6$ GSMS instrument JRC-IRMM has established two complementary techniques for measuring the full isotopic composition of uranium samples.*
*Furthermore, in line with the objectives of the ESARDA Working Group on Techniques and Standards for Destructive Analysis (WG DA), JRC-IRMM is very active in establishing exchange between safeguards authorities, industry and instrument manufacturers. In 2012 JRC-IRMM organised the first $UF_6$ user group meeting. Feedback will be given from the discussions with gas source mass spectrometer users and Thermo Fisher Scientific on relevant issues concerning instrument performance, measurement approaches, needs for further instrument developments and certified reference materials.*

**Keywords:** Uranium isotope measurements, Uranium Hexafluoride, Gas Source Mass Spectrometer, standardisation

## Introduction

The European Commission's Institute for Reference Materials and Measurements (JRC-IRMM) provides policy support in standardization for nuclear security, nuclear safety and nuclear safeguards. This task was originally stated in the Euratom treaty, where the need for isotope standards is explicitly mentioned recognizing the essential role they play in measurements of nuclear materials. For accurate mass spectrometric measurements of nuclear safeguards samples, suitable isotope reference materials are needed to validate measurement procedures and to calibrate multi-collector and ion counting detector systems. There are only three main providers of nuclear reference materials world-

wide. The National Institute of Standards and Technology (NIST) and, since the late 1970s for the nuclear elements uranium and plutonium, the New Brunswick Laboratory (NBL, US-DOE), the Commission d'ETAblissement des Méthodes d'Analyse du Commissariat à l' énergie atomique (CEA/CETAMA) and the European Commission – Joint Research Center - Institute for Reference Materials and Measurements (IRMM). In the European Union, JRC-IRMM is the recognized provider for nuclear isotope reference materials to the nuclear industry and nuclear safeguards authorities.

## Acquisition of a new $UF_6$-Gas Source Mass Spectrometer at JRC-IRMM

Gaseous $UF_6$ is one of the primary forms of uranium in the nuclear fuel cycle, in particular in the isotope enrichment process, which has to be controlled by accurate isotopic measurements of $UF_6$ samples. The direct measurement of $UF_6$ gas has several advantages. Firstly it makes the conversion of $UF_6$ gas into uranium-containing solutions, e.g. by hydrolysis and subsequent treatment, obsolete. Secondly, because of the homogeneity of all gaseous samples, the $UF_6$ gas ion source allows an unprecedented level of reproducibility for isotope ratio measurements. The so-called "double standard" method developed at JRC-IRMM [1, 2], which is a bracketing technique with two isotope reference materials as material standards, ensures that any possible non-linearity or memory effect is eliminated and permits a thorough calculation of the uncertainties on the isotope ratios in the sample. The reproducibility for $n(^{235}U)/n(^{238}U)$ measurements on a $UF_6$ gas source mass spectrometer is at a level of 0.005% RSD (relative standard deviation), which is significantly better (factor of 5) compared with other techniques such as TIMS (Thermal Ionization Mass Spectrometry) and ICP-MS (Inductively Coupled Plasma Mass Spectrometry). In TIMS the reproducibility is still limited even in the most sophisticated modified total evaporation (MTE) measurements [3, 4]. This is due to the fact that the solid sample on a TIMS filament becomes isotopically inhomogeneous during the measurement process and therefore the fractionation history during the measurement is less reproducible, compared to a permanently homogeneous gas sample introduced to a $UF_6$ gas source mass spectrometer. For ICP-MS instruments, the performance for $n(^{235}U)/n(^{238}U)$ ratio measurements is comparable to TIMS, as observed in several inter-laboratory comparison campaigns like e.g. REIMEP 18 [5]. For these reasons the $UF_6$ gas source mass spectrometry technique has been and still remains the reference for uranium isotope measurements at JRC-IRMM.

As an alternative to specially manufactured $UF_6$ gas source mass spectrometers, the isotopic analysis can also be performed using TIMS or ICP-MS instruments after conversion of the $UF_6$ gas into uranyl nitrate solution. This might be advantageous if such instrumentation is already available within the same facility, but it requires additional sample preparation work for the hydrolysis, calcination and dissolution treatment. Thus, unless the TIMS and ICP-MS performance is not significantly superior compared to $UF_6$ GSMS for a specific application, e.g. for measurement of extremely low $^{236}U$ abundances, the $UF_6$ GSMS technique is considered advantageous and less time consuming.

The first GSMS instrument for $UF_6$ isotopic analysis at JRC-IRMM was the VARIAN MAT511 installed in 1972, which has been in full use until recently. This instrument is equipped with a permanent magnet and a double Faraday collector for measuring only the major ratio $n(^{235}U)/n(^{238}U)$, by detecting the most abundant molecular ion species $UF_5^+$, simultaneously at mass 330 (amu) for $^{235}U^{19}F_5^+$ and mass 333 (amu) for $^{238}U^{19}F_5^+$. The instrument has a dual inlet system, which can be used to measure a sample against one standard, usually a certified reference material. Thus for using the DS ("Double Standard") method the sample has to be measured separately against the two standards. The gas pressure and the inflow into the sample container have to be carefully controlled by adjusting the temperature of the sample (or standard) gas ampoule. This is achieved using mixtures of dry ice ($CO_2$) and ethanol; therefore this instrument could only be operated manually. In the late 1980s the successor instrument of the MAT511 came onto the market, the MAT281, manufactured by Finnigan MAT, the successor of VARIAN MAT. This instrument was not installed at JRC-IRMM but at many other $UF_6$ mass spectrometry laboratories. In contrast to the MAT511, the MAT281 is already equipped with a Faraday multi-collector to measure the minor isotopes $^{234}U$ and $^{236}U$ as well, and also with a sample inlet system for up to 9 samples or standards.

In order to replace the almost 40 year old MAT511 $UF_6$ mass spectrometer, a new GSMS instrument called URANUS from Thermo Fisher was installed at JRC-IRMM in 2010 [6]. In contrast to the MAT511, the new URANUS instrument is equipped with a Faraday multi-collector detection system, which allows the minor isotopes $^{234}U$ and $^{236}U$ to be measured simultaneously with the major isotopes

$^{235}U$ and $^{238}U$. It is also equipped with the same new type of Faraday cups used for the TRITON TIMS and NEPTUNE MC-ICPMS, which are deeper and made from graphite in order to reduce scattering of electrons and positive ions in the vicinity of the cups. In contrast to the MAT511, the URANUS is equipped with a sample inlet system for nine samples or standards. This allows automated measurement sequences of several samples using various standards to be performed.

The minor ratio capability of the URANUS is important for JRC-IRMM to facilitate the preparation of tailor-made uranium mixtures on customer request, because no separate TIMS measurements for the minor isotopes have to be performed within the iterative mixing process. The requests on reference measurements of uranium minor isotope ratios in $UF_6$ samples have increased with time because "U-minors" give an indication to safeguards authorities of different feed materials or enrichment processes in enrichment facilities. The importance of the minor ratio capability via $UF_6$ gas source mass spectrometry is to have, in addition to TIMS, another independent technique available for certification and verification at JRC-IRMM of both the major and the minor uranium isotope ratios.

## Performance of the new $UF_6$-Gas Source Mass Spectrometer at JRC-IRMM

Recent detailed investigations [6] have led to new correction algorithms for memory effects and ion scattering to establish SI traceable measurements of both major and minor ratios using the new URANUS $UF_6$ gas source multi-collector mass spectrometer.
For $UF_6$ measurements using the old MAT511 the DS method has been traditionally used, for which a sample is measured against 2 standards, usually certified isotope reference materials, with one of them having a smaller and the other one having a higher $^{235}U$ enrichment compared to the sample to be analysed. The DS method only yields accurate measurement results if memory effects are either negligible or cancelled out, which requires equal "isotopic distances" between the sample and each of the 2 standards. To allow assymmetric "isotopic distances" between the sample and the 2 standards, useful algorithms for memory corrections were derived. The new MCDS ("memory corrected double standard") method is the preferred type of correction because it combines the DS correction with a memory correction derived from the same 2 standards within the same measurement sequence under the same conditions. A more detailed investigation of memory effects for the major ratio revealed a slight (almost insignificant) trend of the memory factor depending on the ratio between the isotope ratios of the samples and/or standards. In order to avoid a large increase of the combined uncertainties of the corrected ratios from the quite large relative uncertainty of this trend (almost 100%), a restriction to a factor of 4 was prescribed between the isotope ratios of the two bracketing standards to be used. Using the MCDS correction type, relative expanded uncertainties of 0.03%-0.04% ($k=2$) are obtained for measurements of the major ratio $n(^{235}U)/n(^{238}U)$. These uncertainties are largely dominated by the uncertainties of the values for the standards used.

For the minor ratio $n(^{234}U)/n(^{238}U)$ the MCDS correction type was found to be the most suitable as well, with relative expanded uncertainties of 0.3%-0.5% ($k$=2). These are obviously not as small as those observed in TIMS/MTE measurements [3, 4], but in many cases fit-for-purpose meeting the customer's requirements.

For the minor ratio $n(^{236}U)/n(^{238}U)$ memory effects play a stronger role compared to the other isotope ratios, and the selection of the optimal correction type depends on the differences between the $n(^{236}U)/n(^{238}U)$ ratios of the sample and the 2 standards [6]. It was determined that the 2 standards have to be selected in a way that the isotope ratio of at least one standard does not deviate by more than one order of magnitude from the sample ratio. For measurements of $n(^{236}U)/n(^{238}U)$ ratios using the URANUS, a detection limit of $3 \times 10^{-6}$ was determined, which can be considered satisfactory, due to the fact that no energy filter is available in front of the detector . However, if the presence of $^{236}U$ in an unknown sample has to be investigated with a lower detection limit, the use of the TIMS/MTE method in combination with a secondary electron multiplier (SEM) for the detection of $^{236}U$ would be preferred, for which detection limits at the order of $6 \times 10^{-10}$ [3, 4] have been reported.

## Outlook

As a result of this thorough validation study for accurate $UF_6$ isotopic measurements using the "URANUS", a careful revision of some of the documentary standards for $UF_6$ gas source mass

spectrometry measurements (e.g. American Society for Testing and Materials (ASTM C1429-99) is now strongly suggested by JRC-IRMM. JRC-IRMM has been invited by ASTM International to develop new and revise existing uranium measurement standards with the ASTM Committee C26 on Nuclear Fuel Cycle. Furthermore the JRC-IRMM advances in reference materials and measurements will be presented at the next ASTM/CETAMA workshop on analytical development and reference materials in the nuclear fuel cycle [7]. As a further consequence of these recent improvements JRC-IRMM emphasises that a larger variety of certified isotopic reference materials with small uncertainties on the certified minor ratios has to be provided, if customers want to perform reliable measurements of the minor ratios by $UF_6$ GSMS. It was therefore decided to use the TIMS/MTE technique to perform a new certification exercise for the basic $UF_6$ reference materials such as the frequently used IRMM-019-029 series.

It was also recognized that an intensified contact with other users of the same instrument, the URANUS from Thermo Fisher, would be very useful. Therefore, in May 2012, the first "$UF_6$ user group meeting" was organized at JRC-IRMM, with participants from Urenco (Gronau/Germany and Almelo/Netherlands), Enritech (Jueliech/Germany), Thermo Fisher (Bremen/Germany), IPEN (Sao Paolo/Brazil) and the IAEA (Seibersdorf/Austria), in line with the objectives of the ESARDA Working Group on Techniques and Standards for Destructive Analysis (WG DA) to facilitate technical exchange between safeguards, industry and instrument developers. It was agreed by all participants that this type of user group meeting should be held regularly e.g. every other year, probably again at IRMM, in order to check if the discussed problems have been resolved and to share new information, experiences with the instruments, new methods, ideas, etc.

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# Improved proliferation resistance of fast reactor blankets manufactured from spent nuclear fuel

*C. Hellesen, S Grape, A Håkansson, S Jacobsson Svärd, P. Jansson*

Division of Appllied Nuclear 'Physics
Department of Physics and Astronomy, Uppsala University
Ångströmlaboratoriet, Lägerhyddsvägen 1, 751 20 Uppsala Sweden
E-mail: carl.hellesen@physics.uu.se

***Abstract:***

In this paper we investigate how a blanket manufactured from recycled light water reactor (LWR) waste, instead of depleted uranium (DU), could potentially improve the non- proliferation characteristics. The blanket made from LWR waste would from the start of operation contain a fraction of plutonium isotopes unsuitable for weapons production. As 239Pu is bred in the blanket it is therefore always mixed with the plutonium already present.

We use a Monte Carlo model of the advanced burner test reactor (ABTR) as reference design, and the proliferation resistance of the blanket material is evaluated for two criteria, spontaneous neutron emission and decay heat. We show that it is possible to achieve a production of plutonium with proliferation resistance comparable to light water reactor waste with a burnup of 50MWd/kg.

**Keywords:** Fertile blankets, proliferation resistance, fast reactor

## 1. Introduction

Today, nuclear power provides about 14% of the world's electricity production from an installed base of almost 400 GW. At this level the supply of fissile material is not likely to be an issue for the foreseeable future. However, it is projected that the world's electricity demand will double by 2050. If nuclear power should make a large sustainable long-term impact in the reduction of green house gases, a system with some form of fissile material breeding is likely to be needed in the future.

For this reason one of the goals of the concept of 4$^{th}$ generation nuclear power (Gen IV) addresses long-term sustainability [1]. In addition, Gen IV should also excel in safety and reliability as well as proliferation resistance. Due to concerns regarding proliferation resistance, the use of fertile blankets is often excluded from the Gen IV concept as it can result in the production of weapons grade plutonium.

However, the use of blankets can provide several benefits. The neutron economy can be significantly improved, simplifying a net breeding of fissile material. An internal fertile blanket is also proposed to be used in the ASTRID project [2] to enhance the passive safety characteristics. Evidently, blankets cannot be ruled out despite the non-proliferation issues, and in this paper we investigate how a blanket manufactured from recycled light water reactor (LWR) waste, instead of depleted uranium (DU), could potentially improve the non-proliferation characteristics.

### 1.1 Generation IV goals

The goals that are set up for the Gen IV system can be seen to be somewhat contradicting. Optimizing the system for one or two goals can have negative impacts on a third; fertile blankets being one example. We summarize it as follows.

The sustainability goal dictates that a Gen IV system should promote long-term availability of nuclear fuel and utilize natural resources in the most efficient way. This translates to a trans uranium (TRU) conversion ratio ($C_R$) of at least 1.0 at the system level. Achieving $C_R > 1.0$ can be done either with high-leakage reactor cores surrounded by fertile blankets or low-leakage homogenous reactor cores.

The safety and reliability goal states that the reactors should have a very low likelihood and degree of core damage, and eliminate the need for offsite emergency response. This goal requires reactors with negative feedbacks that passively shuts the reactor down in the event of, e.g., unprotected loss of flow (LOF) or loss of heat sink (LOHS) accidents. A very illustrative case of this is the EBR-II passive safety tests performed in 1986 [3].

Optimizing the reactor parameters that are important in this case, such as void worth, typically favors small reactor cores with high leakage [4]. For larger low-leakage cores the feedbacks can instead be strongly positive. Compare for example the 1000 $MW_{th}$ Advanced Burner Reactor (ABR) that in $C_R$=1.0 configuration has a total void worth of 6.3$ [5] with the similar but downsized 250 $MW_{th}$ Advanced Test Burner Reactor, which has a total void worth of 1.32$ [6]. The latter would however require radial blankets to reach $C_R = 1.0$

Finally, the proliferation resistance goal states that a Gen IV nuclear power system should be a very unattractive route for diversion or theft of weapons-usable materials. Meeting the non-proliferation goal together with both the sustainability and safety goals will be a challenge.

### 1.2. Reprocessing in generation IV systems

Since enrichment is typically not a part of a Gen IV system the critical step in terms of non-proliferation is the reprocessing where any production of weapons material would take place. The approaches at the system level are either integral or centralized.

In the integral scenario, e.g. the integral fast reactor (IFR), the entire fuel cycle containing reactor, reprocessing systems and fuel fabrication is located in the same building or site. The only fuel and waste transports are the supply of depleted uranium to the site and separated fission products for geological storage from the site. This has been suggested to minimize proliferation risks with regard to theft of fissile material. Any such material is at all times stored either in the reactor or behind heavy radiation shielding, and undetected diversion would be very difficult.

The reprocessing facilities, which for the IFR is a pyroprocessing plant, would also not be capable of separating a pure plutonium product from other trans uranium isotopes (TRU), ensuring that processing of spent fuel never yields a weapons usable product. However, when using a fertile blanket the composition of the fissile material that is bred is essentially weapons grade plutonium (WG-Pu). A pyroprocessing plant, if reconfigured, would be capable of producing a mix with roughly equal proportions of $^{238}U$ and $^{239}Pu$. For weapons purposes this mixture has identical properties to WG-Pu except for a higher critical mass (about 14 kg, compared to 7.4 kg).

Spreading reprocessing technology would therefore pose a proliferation risk. This has led to the centralized approach, supported e.g. by the International Framework for Nuclear Energy Cooperation (IFNEC), where a small number of supplier countries would provide fresh fuel and later take back spent fuel for reprocessing and subsequent recycling. Only the reactor technology would be exported This approach would however lead to shipments of fissile material between supplier and receiver countries, and this material should be as proliferation resistant as possible to minimize the risk from theft. Clearly, reactor designs incorporating fertile blankets would pose a serious proliferation hazard since they contain weapons grade plutonium, and their low burnup provides a minimal radiation barrier for theft and later reprocessing. Any reactor designs incorporating fertile blankets, weather to reach $C_R = 1$ or for safety reasons, should not be considered.

### 1.3. Properties of plutonium in favor of non-proliferation

Two properties of a plutonium product determine its proliferation resistance, the spontaneous neutron emission and the alpha decay heat. A sufficiently strong spontaneous neutron emission would ignite the explosive prematurely and lead to a guaranteed fizzle yield. However, a fizzle would still result in an explosive yield of several 100 tons to a few kiloton of TNT equivalent. Therefore, the international

atomic energy agency (IAEA) considers all plutonium products from nuclear power as a proliferation risk. On the other hand, Kessler [7] proposes that plutonium is proliferation proof if the decay heat is sufficiently large so that either the plutonium core or the surrounding chemical explosives melt, or the chemical explosive self detonate.

Three limiting cases are identified in [7]; a total decay heat larger than 120, 240 and 370 W for a critical mass. These limits correspond to low-, medium- and high-technology weapons manufacturers. The decay heat of $^{239}$Pu is 1.9 W/kg, and a critical mass (7.4 kg[*]) would produce about 14 W in total. On the other hand, the decay heat of $^{238}$Pu is 568 W/kg, and a fraction of less than 3% is enough to result in a total decay heat of 120 W in a critical assembly. $^{238}$Pu is mainly created by alpha decay of $^{242}$Am and is therefore only produced in high burnup fuel. Due to the negligible burnup in blanket assemblies $^{238}$Pu is only present in trace amounts even after irradiation periods of several years.

In this paper we have studied the proliferation resistance of plutonium produced in blanket assemblies where the blankets consist of spent nuclear fuel (SNF) from light water reactors (LWR). Such material typically contains a small amount of proliferation resistant plutonium from the start. The $^{239}$Pu bred in the blanket would therefore always be mixed with the plutonium already present. We investigate using spent nuclear fuel from light water reactors (LWR) as blanket material. This is here referred to as a TRU blanket. LWR spent fuel of high burnup (BU > 50 MWd/kg) is mainly composed of $^{238}$U and about 1-2% of plutonium with properties that are highly unsuitable for weapons purposes.

The concept is similar to that proposed in [8] where a fast reactor blanket spiked with 4% of $^{241}$Am was studied. After neutron capture the americium would decay to $^{238}$Pu, and the $^{239}$Pu produced in the blanket would be mixed with enough $^{238}$Pu to render it proliferation proof. This would require reprocessing steps involving separation of pure americium.

In contrast, using LWR spent fuel as blanket material in e.g. a metal fuel sodium cooled fast reactor (SFR) would only require an oxide reduction step to convert it to metallic form. This oxide reduction step is already a part of the proposed pyroprocessing facility for recycling of LWR oxide fuel waste, so the changes to the proposed fuel cycle are minimal

## 3. Reactor model

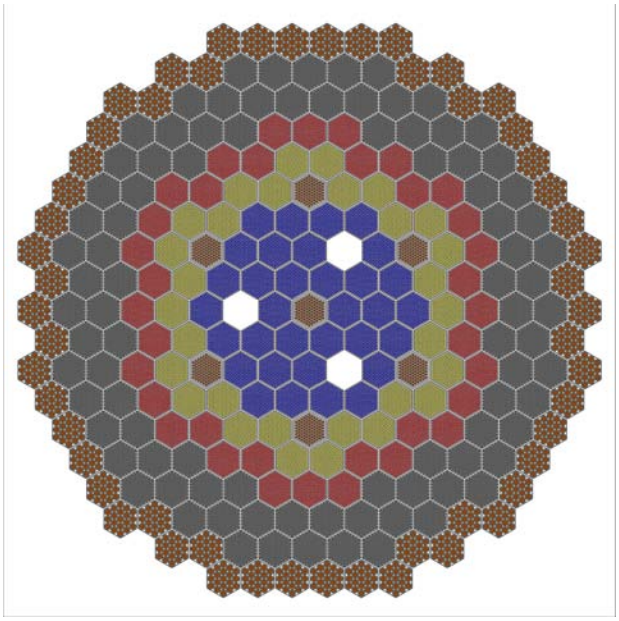


**Figure 1.** Illustration of the SERPENT model of the ABTR used in this paper.

[*] This assumes a plutonium sphere surrounded by a natural uranium reflector of 5 cm.

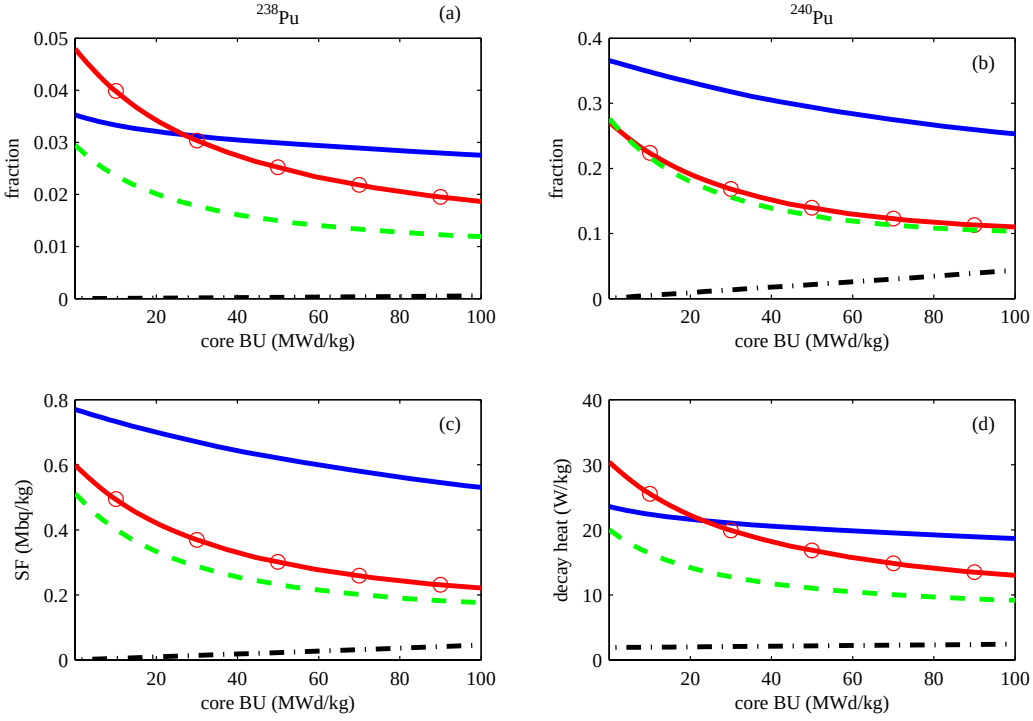
The Advanced Burner Test Reactor (ABTR) developed at Argonne Laboratories [6] is used as our reference reactor model in this study. The ABTR is a sodium cooled metallic fuel design, and in its standard configuration it features a blanket-free core with a TRU conversion ratio of $C_R = 0.65$. It consists of 33 inner and 30 outer assemblies as well as 126 reflector and 54 shield assemblies. An option is also proposed in [6] where the innermost row of reflectors (36 assemblies) is exchanged for a fertile blanket to allow for a conversion ratio $C_R = 1.0$, i.e., a breeder configuration. The driver assemblies contain 217 fuel pins each, while the reflector and shield assemblies contain 91 steel pins and 19 boron carbide ($B_4C$) pins, respectively.

| | DU | UOX 50 MWd/kg | UOX 70 MWd/kg | MOX 50 MWd/kg |
|---|---|---|---|---|
| $^{238}U$ | 99.7 | 97.6 | 97.2 | 93.0 |
| $^{238}Pu$ | 0.00 | 0.03 | 0.07 | 0.2 |
| $^{239}Pu$ | 0.00 | 0.57 | 0.61 | 1.8 |
| $^{240}Pu$ | 0.00 | 0.32 | 0.37 | 2.1 |
| $^{241}Pu$ | 0.00 | 0.13 | 0.16 | 0.6 |
| $^{242}Pu$ | 0.00 | 0.10 | 0.16 | 1.0 |

**Table 1:** Isotopic composition in % for the four different blanket types studied in this paper.

Here we study the blanket version of the ABTR using the Monte Carlo (MC) code SERPENT [9], and the geometry of the model is shown in Figure 1. Four different actinide compositions of the blanket were considered: depleted uranium as a reference scenario and TRU blankets with an actinide mix from spent LWR fuel with 50 and 70 MWd/kg burnup as well as the actinide mix from spent LWR MOX fuel with a burnup of 50 MWd-kg. The isotopic composition of the blanket assemblies, aside of minor actinides, is given in Table 1.

## 4. Results




(a and b) Relative fraction of $^{238}$Pu and $^{240}$Pu in the plutonium bred in blanket assemblies. (c) Spontaneous neutron emission per kg blanket Pu. (d) Decay heat per kg blanket Pu. For all plots, DU is shown as dash-dot black, 50 MWd/kg is shown as dash green, 70 MWd/kg is shown as solid-red + circles and MOX 50MWd/kg is shown as solid blue.

The relative fractions of $^{238}$Pu and $^{241}$Pu for a blanket assembly is shown in Figure 2 (a and b) over an entire ABTR cycle, i.e. reaching a final core burnup of 100 MWd/kg. For All TRU blankets the relative fractions of $^{238}$Pu and $^{241}$Pu show clear decreasing trends that flattens out at the end of the cycle. This is due to more $^{239}$Pu being bred in the blanket while the other Pu isotopes remain at roughly constant levels. The $^{238}$Pu fraction in the DU blanket is negligible at all times.

In Figure 2 (c and d) the total spontaneous neutron rate per kg Pu (SF) as well as decay heat per kg Pu is shown for the different blanket types. The values for all TRU blankets are falling since mainly $^{239}$Pu is produced while the DU blanket shows slightly increasing decay heat and neutron rates since the $^{239}$Pu is gradually being diluted with higher heavier Pu isotopes.

| material type | $f_{\text{fissile}}$ | $M_{\text{crit}}$ [kg] | $n_{\text{spont}}$ [Mbq] | $P_{\text{decay}}$ [W] |
|---|---|---|---|---|
| DU | 0.96 | 7.4 | 0.3 | 18 |
| TRU 50 | 0.86 | 8.0 | 1.4 | 69 |
| TRU 70 | 0.84 | 8.3 | 1.8 | 110 |
| TRU MOX | 0.60 | 10.4 | 5.5 | 195 |
| SNF 50 MWd/kg | 0.61 | 10.4 | 5.3 | 190 |

**Table 2.** Total fissile fraction of the different blanket types at the end of the ABTR cycle (core burnup of 100 MWd/kg) as well as the critical mass. The spontaneous neutron emission and decay heat are given for a critical assembly. Corresponding values for the Pu mix from spent LWR fuel at burnup 50 MWd/kg are given for comparison.

In Table 2 we show details of the plutonium properties at the end of the 100 MWd/kg ABR cycle. The corresponding values for spent LWR fuel is also given for comparison. The total fissile fraction of the plutonium ($^{239}$Pu + $^{241}$Pu) is highest for the DU blanket and progressively lower as the burnup of the blanket material increases. We also show the critical mass of the different plutonium compositions

assuming a spherical core surrounded by a 5 cm thick reflector of natural uranium. This gives the total spontaneous neutron rate and decay heat for a critical assembly.

It can be noted that there is a difference in the composition of the fissile isotopes in Table 2. The fissile part of the DU blanket plutonium is mainly composed of $^{239}$Pu; for the 3 TRU blankets the plutonium also includes 3% to 6% $^{241}$Pu; for spent LWR fuel the plutonium includes about 14% $^{241}$Pu.

## 5. Discussion and conclusions

In this paper we have studied the proliferation resistance of plutonium bred in a fast reactor blanket manufactured from LWR spent fuel (TRU blanket). Two main properties of the resulting plutonium are considered, spontaneous neutron emission and decay heat. We note that for both decay heat and spontaneous neutron emission the situation is, from a non-proliferation point of view, improved considerably compared to a depleted uranium blanket. Using TRU blankets it is possible to achieve a plutonium product with similar proliferation resistance as plutonium in spent LWR fuel with a burnup of 50 MWd/kg.

The level of proliferation resistance of the TRU blanket plutonium is highly dependent on initial LWR burnup. Using spent LWR fuel at 50 MWd/kg as blanket material results in a plutonium product that cannot be considered proliferation proof even for the lowest technology weapons producer. Compared to plutonium from a DU blanket the decay heat for a critical assembly is 4 times higher. This would make manufacturing considerably more difficult, but probably still possible since it falls below the 120 W limit proposed by Kessler [7]. Also, according to [7] we see that a spontaneous neutron rate of 3 Mbq for a total assembly is sufficient to guarantee a minimum fizzle yield of a simple implosion type weapon. A TRU blanket using using spent LWR fuel with a burnup of 50 MWd/kg has a neutron rate of 1.4 Mbq and would result in a highly unpredictable yield, although not a guaranteed fizzle.

On the other hand, using the TRU MOX blanket results in a proliferation resistant plutonium on a par with spent LWR fuel with burnup of 50 MWd/kg. An alpha decay heat of 195 W for a critical assembly would render low weapons technology impossible and medium technology on the borderline. Further, a spontaneous neutron rate of 5.5 Mbq would guarantee a minimum fizzle yield for an implosion design [7].

Finally, the proliferation resistance of the TRU blanket assemblies is highest at the start of the power cycle and gradually decreases as more $^{239}$Pu builds up. This can be considered as a positive feature since it eliminates the risk that a country purchases fresh fuel and only runs a short power cycle, and thereby acquires a very attractive plutonium composition for weapons manufacturing.

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# Confirmation of Nuclear Treaty Limited Items: Pre-dismantlement vs. Post-dismantlement

**Duncan MacArthur, Danielle Hauck, Morag Smith**

Los Alamos National Laboratory
Los Alamos, New Mexico 87544 USA
E-mail: dmacarthur@lanl.gov, hauck@lanl.gov, mks@lanl.gov

***Abstract:***

*One of the key factors in verification of future nuclear disarmament treaties will be the confirmation, by a monitoring party, that declared treaty limited items (TLIs) are consistent with the declaration made by the host country. A significant part of this confirmation is supplied by a radiation measurement system that confirms the declared radiation characteristics of the TLI. These radiation measurements can take the form of measuring declared attributes of the TLI, comparing declared TLIs with a pre-existing template, or some combination of the two techniques. Treaties including TLI dismantlement form an important subset of general disarmament treaties. In a dismantlement scenario, the required radiation measurements can be performed either before or after the TLI is dismantled (or at both times). Pre-dismantlement measurement may give additional confidence that the item is truly a TLI but may be technically challenging while post-dismantlement measurement can offer additional confidence that the item has been truly dismantled. Since repeated measurement increases monitor confidence and there are technical advantages to both measurement times, a combination of pre-dismantlement and post-dismantlement measurement will lead to the highest overall confidence. The relative importance of the two types of measurement is directly dependent on the specifics of the treaty under discussion.*

**Keywords:** nuclear; disarmament; attribute; template; dismantlement

## 1. Nuclear Arms Reduction Treaties

Most treaty monitoring scenarios can be reduced to two requirements, one for the host party and one for the monitoring party:

- The owner of nuclear material or a device (the host party) makes a declaration concerning that item (the treaty-limited item or TLI) and/or its disposition to another entity (the monitoring party).
- The monitoring party must confirm this declaration without observing any sensitive information. By sensitive information, we mean classified information that the two parties do not intend to share.

The crux of the treaty-monitoring measurement challenge lies with the phrase “without observing any sensitive information.” Traditional nondestructive assay (NDA) techniques (based on gamma-ray detection, neutron detection, or calorimetry) are widely and successfully used in numerous scenarios (e.g., waste assay and spent fuel monitoring) that do not involve classified information (unlike treaty monitoring). [1]

Nuclear arms reduction treaties in force today are generally based on the number of delivery vehicles. [2] The use of radiation detection in verification of these treaties is limited to confirming that items are not weapons—lack of radiation levels above background is taken as evidence that an item is not a nuclear weapon. However, it is possible that future monitoring regimes will include warhead confirmation and/or monitored dismantlement. In both cases, it is necessary to confirm that a declared TLI is indeed a warhead. In the remainder of this paper, we address some of the issues surrounding warhead confirmation in a monitored dismantlement scenario.

The host party is responsible for certification of the measurement system and monitoring regime. This certification will include **information certification** (that sensitive information will not be divulged), **facility certification** (that all facility safety and security regulations have been met), and **nuclear explosive safety certification** (that the measurement system and process are acceptable for use on the host's declared TLIs). The certification challenges associated with allowing a monitor to confirm that an item is a warhead are much more complex than those associated with confirming that an item is not a warhead. In particular, warhead confirmation involves radiation measurements on sensitive nuclear items, and identifying defining characteristics of these items--both of which could reveal sensitive information.

Similarly, the monitoring party is responsible for all steps required to build monitoring party confidence in the measurement system and its use within the monitoring regime; a process generally termed authentication. Regime authentication will include consideration of **measurement system authentication** (that the measurement system is making the agreed upon measurements) and **protocol and procedures authentication** (that the treaty protocol and procedures allow for independent confirmation of the host's declaration). Other authentication concerns include the **context and environment** in which the system will be operated (discussed briefly below in section 2.1), and maintaining knowledge of location (termed chain of custody or **CoC**) of the TLI, measurement system, and calibration sources. The additional limitations on the monitoring regime that are prompted by the host's certification of warhead measurements make it more difficult for the monitoring party to maintain confidence in the monitoring regime and its results.

Arms reduction treaties are often reciprocal and each party must verify each other's declarations. Thus, each country will play the role of both host and monitor during the course of the treaty. In particular, each authentication or certification technique proposed for a given host/monitor pairing must be evaluated by both countries for its impact when the roles are reversed.

Although many of the specific examples in the remainder of this paper refer to the monitored dismantlement scenario, the concepts, and in particular the measurement concepts, apply more generally to treaties involving warhead (as opposed to delivery vehicle) confirmation. In sections 2 and 3, we will review dismantlement and measurement concepts while in section 4 we move on to a discussion of the timing of confirmation measurements.

## 2. Dismantlement Treaty Verification

As noted above, treaty verification generally involves confirmation of a TLI declaration or confirmation of a disposition declaration. Within a monitored warhead dismantlement treaty, these become warhead confirmation and dismantlement confirmation.

For the purposes of this discussion, we will define dismantlement of a nuclear weapon as the separation of fissile material (FM) from high explosive (HE). Given this definition, dismantlement confirmation can be achieved by demonstrating FM presence and HE absence in the declared FM container along with confirming that there is no FM in other containers or remaining in the dismantlement area. If HE were tracked, similar confidence could be achieved by confirming presence of HE and absence of FM in the declared HE container and no HE elsewhere—however, "no HE elsewhere" is very difficult to confirm practically. It is easier to detect and track undeclared FM in containers and large areas. Dismantlement confirmation, regardless of method, obviously must occur "post-dismantlement."

As described above, a combination of presence and absence measurements can be used to confirm that a nuclear item has been dismantled. Confirming that that a declared item is a nuclear warhead is the more difficult to the two problems. As shown in Figure 1, there are four points at which confirmation measurements might be performed on a declared warhead: (1) upon entering the monitoring regime, (2) somewhere within the CoC regime, (3) immediately prior to dismantlement, and (4) immediately after dismantlement. We will discuss the options for when to perform warhead confirmation measurements and the influence of other aspects of the monitoring regime (such as CoC) on these timing choices (and *vice-versa*). Warhead confirmation measurements are important not only to confirm that the item being monitored is indeed a warhead, but also that the item that is dismantled was indeed a warhead.

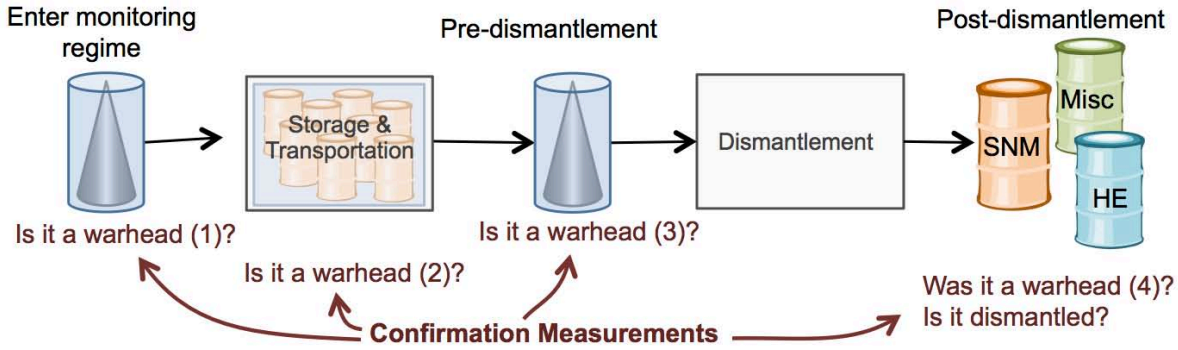


Figure 1. Schematic representation of a possible dismantlement regime identifying potential confirmation measurements. The dismantlement process is shown as a black box as it is anticipated that the host country will consider the details of this process sensitive.

## 3. Confirmatory Measurements

We have identified three primary methods to warhead confirmation. These are:

• **Attribute Measurements**: Are specified item properties consistent with it being a warhead?
• **Template comparison**: Is the item consistent with other items known or believed to be warheads?
• **Provenance**: Has the item undergone movements or come from a location consistent with being a warhead?

The first two methods are measurement-based and are described in detail below. The third approach is to use the provenance of the item as evidence that it is a warhead, and to maintain continuity of knowledge on the item through the remainder of the monitoring regime. Even if any one of these methods produces relatively low assurance, all three can be used in combination to increase monitor confidence. The host's definition of "sensitive information" limits all three techniques, but each is limited in a different way.

In this paper, we focus on measurement methods (both attribute and template) and mention provenance only in passing, even though provenance may be an important source of confidence and can be used in conjunction with the warhead confirmation measurements discussed here. [3]

### 3.1. Information Barriers

Confirmatory measurements often involve the collection of sensitive data. The measurement system must report the non-sensitive results, while simultaneously protecting any intermediate data required for the measurements. A key component of confirmation measurement systems is the information barrier (IB). [4] The IB is a series of controls that ensures that no sensitive information is released during a measurement and, simultaneously, that the monitoring party is able to independently confirm the host's declaration concerning the measured TLI.

A conceptual drawing of a generic measurement system incorporating an IB is shown in Figure 2. In practice, an IB would not be the single shell shown below—a practical IB includes layers of hardware, software, and procedural protection to provide a barrier system that, as a whole, are fault resistant and the components of which are fault tolerant.

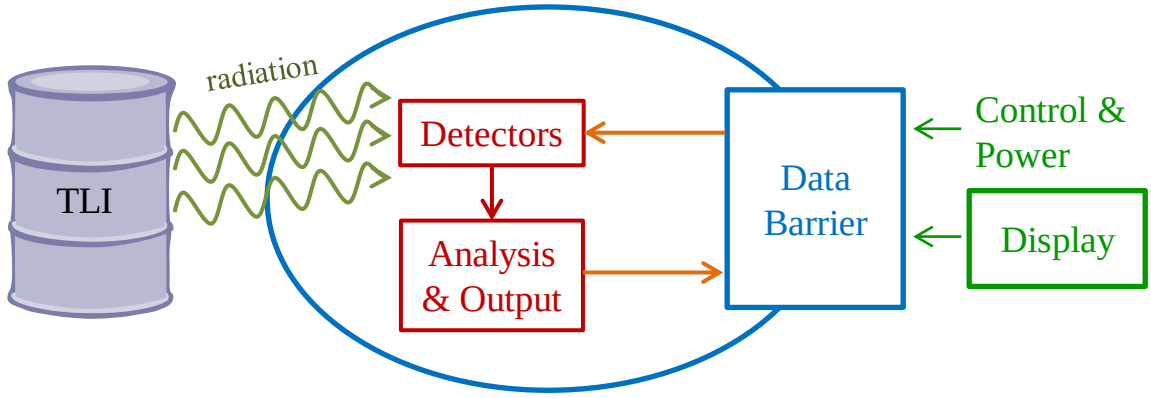


Figure 2. Schematic representation of an information barrier (IB) for a generic measurement system. All potentially sensitive information (contained in parts of the system shown in red) is contained within the IB (blue) while the monitors (green) are outside of the IB. In an attribute measurement system the IB prevents the release of any information other than the attribute results. In a template comparison system, the IB prevents the release of any information other than positive or negative results of a successful comparison.

Unfortunately, the same IB system that excels at protecting the host party's classified information also excels at "protecting" the monitoring party from any information that could be used to confirm the host party's declaration. Attribute measurement (discussed in section 3.2) and template comparison (discussed in section 3.3) are both ways of presenting useful non-sensitive results based on sensitive data.

### 3.2. Attribute Measurement

One approach to confirming that an item is consistent with a warhead with a carefully controlled release of information about the TLI is to use an attribute measurement system (AMS). Attributes, as defined here, are unclassified indicators of potentially sensitive measurement results. Potentially sensitive information can be made into an attribute by comparing the information with a threshold, e.g., the attribute is "quantity above threshold." Some potential attributes are:

- the presence of nuclear material,
- having a nuclear material mass above a threshold,
- having a plutonium isotopic ratio below a threshold, or
- having a uranium enrichment above a threshold.

In any fielded implementation of an AMS, the host and monitoring parties would agree on the attributes to be measured (as well as the details of the AMS itself). The confidence generated by an AMS is only as high as the confidence that the chosen attributes uniquely define a warhead. The choice of attributes is very important—not only must the attribute display be non-sensitive, but the reason for choosing that attribute must also be non-sensitive. Thus, negotiated attributes are often bounded by sensitivity concerns and may not be capable of providing a high level of confidence in warhead identity.

A number of AMSs have been built and demonstrated for international audiences in the last 15 years. Three significant examples are:

- *Trilateral Initiative Demonstration system* - designed and built in the U.S., measured 3 attributes, demonstrated for IAEA and Russian representatives, focus on information barrier capability. [5]
- *Fissile Material Transparency Technology Demonstration* (FMTTD) – designed and built in the U.S., measured six attributes; demonstrated to Russian and U.S. government representatives, focus on certification and information security. This is the only AMS where a classified weapon component was measured in front of an uncleared audience. [6]
- *Attribute Verification system for Neutrons and Gammas* (AVNG) - trilaterally designed (VNIIEF, IAEA, LANL/LLNL), jointly developed (VNIIEF, LANL/LLNL) and built in Russia, measured three attributes, demonstrated for a U.S. audience, focus was on certification. [7]

An AMS performs independent measurements on each item. Thus, the confirmation (or lack thereof) of each declared item is completely independent of measurements on other items. Since each measurement stands alone, no long-term storage of classified information is required. As long as the same attributes are declared, the measurement system can be used with several types of TLI or the same TLI in different containers. Finally, there can be a good match between the declared characteristics of the TLI and what is actually measured.

### 3.3. Template Comparison

Another approach to the challenge of generating monitor confidence is to use template matching. In this case, a potentially sensitive signature (most often a radiation signature) from one declared TLI is compared with a similar signature from an item known to be a TLI. This comparison can generate a high level of confidence that two items are identical or that a given item is unchanged. Since the template and individual results are not shown to the monitoring party, the template itself can contain sensitive information.

Several template-matching demonstrations occurred over the same 15-year time frame as the AMS development discussed above. Of note are:

- *Trusted Radiation Identification System* (TRIS) that was a U.S. developed system designed to provide a means to use low-resolution gamma-ray spectral measurements from sodium iodide (NaI) detectors to confirm the identities of declared material. TRIS compares the radiation signature of an inspected item with a known standard for a weapon or component of the same type. [8]
- A *template-matching demonstration* with classified canned subassemblies in containers was held in the U.S. in 1999. In this demonstration reference signatures were acquired for two containers with different items with the Russian delegation present. The signature was obtained for a third item in the third container and it was shown to match one of the reference signatures. This comparison was displayed on the computer screen for viewing by a Russian delegation with the ordinate scrambled.
- A similar *template-matching demonstration* was performed with three classified plutonium metal parts in containers at VNIIEF for a US delegation with the same display of the ratio of signatures with the ordinate scrambled.

In a template comparison, two items can be compared without ever releasing the template itself. The major advantage of this is that a template can incorporate a broad range of potentially sensitive radiation signatures (or other item features) and can result in high confidence that two items are nominally identical. However, template comparisons require long-term jointly controlled storage of sensitive information. The information barrier in this scenario must include a robust methodology for storing and comparing sensitive information without unintended release.

Since templates can result in high confidence that two items are nominally identical, templates have a large potential role in maintaining CoC. [9] However, for the purpose of warhead confirmation, confidence in a template comparison is only as high as confidence that the comparison copy is legitimate. In addition, whereas attribute measurements result in independent confidence levels for each warhead, template comparisons result in correlated confidence levels for each item. Confidence (or lack thereof) in the legitimacy of the comparison copy automatically transfers to the level of confidence in the accuracy of warhead confirmation for an entire series of items.

## 4. Timing of Warhead Confirmation Measurements

As illustrated in Figure 1, warhead confirmatory measurement can be performed at four different times within the dismantlement process. As described below, each of these times has specific advantages and disadvantages. The different approaches to warhead confirmation (templates, attributes, provenancing or a combination thereof) offer different levels of confidence at different times; thus, the details of a warhead confirmation measurement will influence optimum timing of that measurement.

The timing of a confirmation measurement can also result in a trade-off between measurement complexity and CoC complexity. The availability of CoC tools and item provenance must be taken into account when determining the optimum times to perform warhead confirmation measurements. CoC can be maintained during storage and transportation using a combination of visual observation and tags and seals. Maintaining CoC through the dismantlement process requires more elaborate measures such as the "room within a room" discussed in another paper at this conference. [10]

### 4.1. Pre-dismantlement – Entry into Monitoring Regime

We define entry into the monitoring regime as the time when the monitoring party has the option to begin maintaining CoC on the item. We do not assume that warhead confirmation measurements are necessary to "initialize" an item into the monitoring regime but that this entry time is one potential time to perform confirmation measurements. Confirmation measurements could be based on attribute measurement or template comparison or both.

Performing confirmation measurements upon entry into the monitoring regime provides the monitor with immediate confidence that an item is as declared. Otherwise, an item may be present (and potentially tracked) within the monitoring regime for many years before achieving confidence that it is a TLI. In addition, if the item has a known useful provenance (in this case, useful means that the provenance provides evidence that the item is a warhead), then immediate warhead confirmation measurements together with the provenance may be the best way to provide strong confidence that the item is as declared.

The timing of the warhead confirmation measurements has ramifications for the importance of maintaining CoC. If confirmation upon entry into the monitoring regime is the only warhead confirmation measurement prior to dismantlement, then CoC between entry into the regime and dismantlement is extremely important. If, on the other hand, the item does not have a useful provenance and if the movements within the regime are not useful for confirming that the item is a warhead, then it may not be important to perform warhead confirmation measurements at entry into the monitoring regime; in this case, the importance of CoC at any time prior to the first warhead confirmation measurements is minimized.

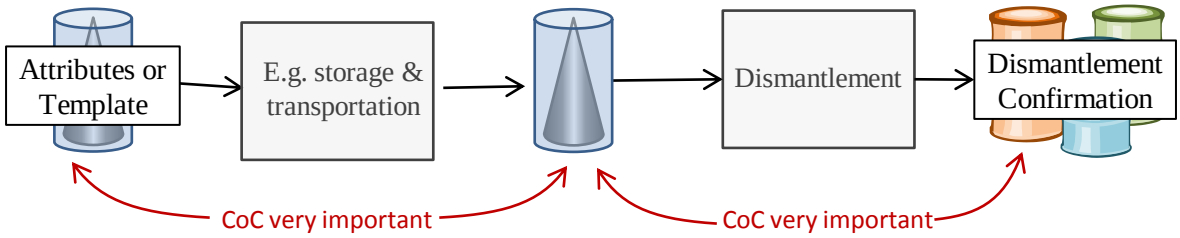


**Figure 3. Ramification on CoC for performing warhead confirmation measurements upon entry into the monitoring regime**. If there is only one set of warhead confirmation measurements, CoC between those measurements and dismantlement confirmation is extremely important. If the confirmation measurements are performed upon entry into the monitoring regime, there may be a relatively long amount of time (up to years) between the measurements and the dismantlement.

Confirmation made upon entry into the regime has the strongest tie to warhead provenance but requires long-term CoC within the regime to connect the confirmation with eventual dismantlement, which may occur many years later. In addition, confirmation at this time involves the relatively difficult technical challenge of measuring assembled weapons where nuclear signatures may be shielded by explosive material and/or the casing of the weapon itself. Information security concerns may also be heightened for measurements of an assembled weapon.

### 4.2. Pre-dismantlement – Immediately Pre-dismantlement

Another potential time for warhead confirmation measurements is immediately prior to dismantlement. Such a measurement may or may not represent the first set of confirmation measurements made on the declared TLI. Confirmation measurements made immediately prior to dismantlement provide added confidence that the item entering the dismantlement process is truly a warhead. The more time

that elapses between the most recent warhead confirmation measurement and the dismantlement process, the more difficult it will be to maintain CoC.

If the confirmation measurements are only performed immediately pre-dismantlement, then maintaining CoC prior to this first set of measurements is only valuable if using provenance and/or movement of the item is a source of confidence in item legitimacy.

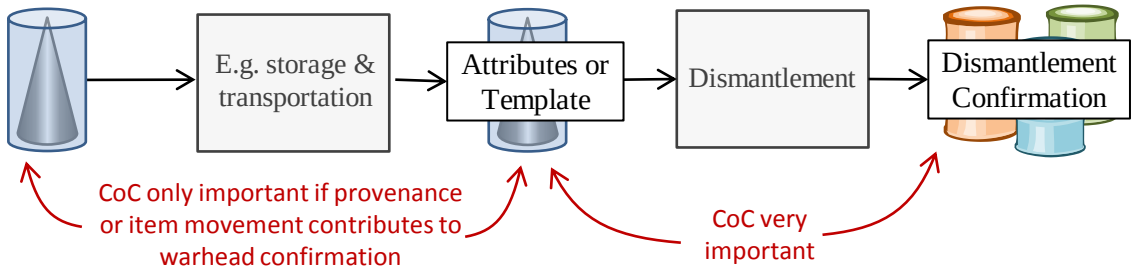


**Figure 3. Ramifications on CoC for attribute measurements pre-dismantlement only.** If the provenance or item movement is a source of confidence, then CoC prior to the first warhead measurement is very important. If provenance and item movement do not provide confidence, then the value of maintaining CoC prior to confirmation measurements is questionable.

If confirmation is made immediately prior to dismantlement, the tie to dismantlement is much stronger, but the link to the original declaration, and entry into the regime, requires extensive CoC. As the warhead is still assembled at this point in time, the measurement challenge is probably no different than in section 4.1.

### 4.3. Pre-dismantlement – During transportation and storage

Warhead confirmation could also occur at any time during the pre-dismantlement storage and transportation. For the most part, the strengths and trade-offs inherent in this measurement time fall on a sliding scale between the “Entry into Regime” described in section 4.1 and the “Immediately Pre-dismantlement” described in section 4.2. However, the one significant exception is that a robust CoC connection is now required for linking to both end points—the direct tie to either provenance or dismantlement has been lost with no specific compensating gain. For this reason, all else being equal, we would advocate making the primary confirmation measurements at one end or the other, and not at an intermediate position.

However, all else is seldom equal. If the CoC regime can begin on the delivery vehicle itself, practical and access limitations may prevent the confirmation measurement from being made simultaneously. In this case, performing a confirmation measurement as soon as possible following entry into the CoC regime may increase monitor confidence.

In addition, random “challenge” measurements during transportation and storage can increase the monitoring party’s confidence that the TLI remains as declared throughout the process. These intermediate measurements would not replace the confirmation measurements, but would increase confidence that the item being tracked is a TLI. Even if attribute measurements and/or provenance are used for initial confirmation, a template comparison may be the most effective way to perform these challenge measurements.

Another use of “intermediate” confirmation measurements is to re-establish CoC on an item if CoC has been lost at some point during the transportation and storage operations. It is never the intention to lose CoC, but it is important to have a recovery mechanism, such as re-confirmation, in case such a loss occurs. In addition, the access constraints discussed further in section 4.5 may have a significant impact on confirmation timing.

### 4.4. Post-dismantlement

There are two potential types of confirmation measurement that may take place post-dismantlement. The first, and most obvious, is to confirm that an item is dismantled, i.e., that the HE and the FM are

separate. Although the dismantlement confirmation itself presents challenges (as touched upon in section 2), the timing of dismantlement confirmation is not in doubt.

Following dismantlement, the TLI confirmation measurement is significantly changed and may be simplified. However, some of the critical characteristics that "make a weapon a weapon" may be lost in the dismantlement process:

- The shielding effects of HE may be reduced or eliminated in a post-dismantlement measurement. In addition, since the FM is no longer part of a warhead, the storage and packaging requirements will be changed. If the relevant characteristics are primarily nuclear, like the attribute examples given above, then the reduction in shielding may make nuclear measurements faster, more effective, and more discriminating.
- However, if other characteristics, such as relative FM and HE geometry, are important in the definition of a warhead, then post-dismantlement confirmation adds little or no confidence to an attribute measurement. The effectiveness of this type of warhead confirmation depends explicitly of the declared characteristics of the warhead and the mix of attribute measurement, template comparison and provenance used to make this confirmation.

The CoC requirements for confirmation that a declared item has been dismantled are essentially non-existent as warhead confirmation and dismantlement confirmation are occurring simultaneously (or nearly so). Conversely the CoC requirements for tying the dismantled TLI to the originally declared TLI become more extreme—in particular, the link now passes though the dismantlement "black box."

### 4.5. Comparisons and Analysis

All four potential times for performing warhead confirmation measurements illustrated in Figure 1 have advantages and disadvantages—some of which have been discussed above. The determination of which confirmation timing is most suitable will depend directly on the details of the particular treaty being confirmed. Two limiting examples can illustrate this concept:

- If a treaty is purely concerned with item dismantlement, then post-dismantlement confirmation provides the strongest link between the item and the dismantlement process. In this extreme, CoC prior to dismantlement becomes less important as the link to regime entry is not a major goal.
- Conversely, if a treaty is purely concerned with keeping track of warheads within a monitoring regime, then confirmation upon entry into the regime provides the strongest, and most timely, tie to the warheads themselves.

In practice, it seems unlikely that a monitored dismantlement treaty would fall into either of these extreme cases. Some of the factors used in determining the most effective times to perform warhead confirmation measurements are: (1) the treaty importance of accepting warheads into the regime, (2) whether a useful provenance of the warhead is available, (3) whether the movements of the item through the monitoring regime provide additional confidence that the item is a warhead, (4) the relative confidence in CoC during different stages of the monitoring regime, (5) the planned types of warhead confirmation measurements (templates or attributes), (6) practical considerations such as the measurement difficulty due to amount of shielding around the item at different points in the monitoring regime, (7) the degree of host sensitivity concerning the container details and/or geometry of the item at different points in the monitoring regime, (8) safety requirements for measuring the item at different points in the monitoring regime, and (9) accessibility of the item, measurement equipment and/or radiation test sources at different points in the monitoring regime.

As *regime acceptance (1)* becomes more important in treaty verification, warhead confirmation at entry into the monitoring regime also becomes more favoured. If *item provenance (2)* is used as a source of confidence that the dismantled item was indeed a warhead, then it is necessary to maintain CoC continuously from entry into the monitoring regime to post dismantlement regardless of when warhead confirmation measurements are performed. Similarly, if *item movements (3)* through the monitoring regime add confidence in item legitimacy, then CoC must be maintained continuously starting before the movements until after dismantlement. The optimum timing of confirmation measurements may still be influenced by confidence in CoC (as discussed below) but may be dominated by considerations of measurement ability, safety and security.

If item provenance and movements provide only limited confidence and must be supplemented with confirmation measurements, then the timing of warhead confirmation greatly influences (and is influenced by) the *type of CoC (4)* required during different phases of the monitoring regime. The availability of CoC tools influences the optimum timing of warhead confirmation measurements, and maintaining CoC during dismantlement is relatively more difficult than maintaining CoC during storage and transportation.

A good example of a trade-off between CoC and confirmation timing results from the differences of performing confirmation measurements either before or after dismantlement. Regardless of timing, maintaining CoC between the most recent warhead confirmation measurements and post-dismantlement is necessary in order to confirm that the item that is dismantled does (or did) meet warhead confirmation criteria. If the warhead confirmation measurements can be performed prior to dismantlement, then CoC must be maintained through the dismantlement process. Performing warhead confirmation measurements post-dismantlement avoids the relatively difficult task of maintaining CoC during dismantlement.

There is interplay between the timing of different *types of warhead confirmation measurements (5)*. In particular, a template comparison may be best performed prior to dismantlement due to the changes in radiation signatures that accompany dismantlement. Although the attribute examples given in this paper can be measured either before or after dismantlement, some other potential attributes, such as ones based on relative geometry of FM and HE, could only be performed prior to dismantlement. Attributes that must, by their nature, be performed prior to dismantlement may provide a stronger indication that an item is a warhead. However, such attributes could also be more sensitive and have not been used in measurement systems to date.

There are *practical (6)* and *security (7)* ramifications of measuring assembled weapons pre-dismantlement or components post-dismantlement. Prior to dismantlement, the assembled item may have more shielding, thus complicating the technical ability to make the measurement and the geometry of the assembled weapon may be more sensitive than the geometry of the disassembled components increasing security concerns. The *safety (8)* ramifications depend on the item and the facility. An assembled weapon is usually in a highly stable (and safe) state and it may be easier to perform confirmation measurements on an assembled weapon than on a disassembled component containing HE. On the other hand, confirmation measurements on a disassembled FM component may have fewer safety considerations than measurements on an assembled system containing HE. Safety considerations will influence access (and in particular standoff distance) to the item for the measurement equipment.

The *(9) accessibility* of an item through the various stages of monitored dismantlement can have a very direct influence on the optimum confirmation timing. Measurements such as neutron multiplicity counting and image generation can require large detectors, which are physically incompatible with some locations. As another example, dedicated facilities may have fewer sensitive characteristics, making it easier for monitors to move around the facility and perform necessary measurements.

Perhaps the best combination (from a monitoring point of view) of measurement strengths, CoC strengths, and efficiency would be achieved by doing warhead confirmation upon entry into the regime (or as early as practical), immediately pre-dismantlement, and post-dismantlement, with a monitoring party option to perform confirmation measurements at various points during the storage and transportation phases. Steps back from this ideal would be made after considering the nine factors described above. One possible way to decrease the measurement burden while maintaining monitor confidence would be to give the monitoring party the option of performing measurements at any of these times while still requiring that the confirmation measurements be performed a certain percentage of the time.

## 5. Acknowledgements

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# Prototype Development and Field Trials under the Next Generation Safeguards Initiative Spent Fuel Non-Destructive Assay Project

**S.J. Tobin[1], H.O. Menlove[1], M.T. Swinhoe[1], H. Trellue[1], M. Humphrey[2], A. Belian[1], T. Burr[1], L.W. Campbell[3], W. Charlton[4], J.M. Eigenbrodt[1,4], A. Favalli[1], N. Fischer[13], J. Galloway[1], I. Gauld[5], J. Gerhart[1,4], S. Grape[6],Y. Ham[7], J. Hendricks[1], V. Henzel[1], D. Henzlova[1], J. Hu[5], A. Hunt[8], P. Jansson[6], A. Kaplan[9], H.D. Kim[10], D. Kim[10], I.C. Kwon[10], J. Kulisek[3], H. Liljenfeldt[11], N. Lundkivist[1,6], A.M. LaFleur[1], T.P. Lou[12], B.A. Ludewigt[12], V. Mozin[7], J.M. Oh[10], S.H. Park[10],S.A. Pozzi[9], D. Rodriguez[3], C. Romano[5], H.S. Shin[10],T.J. Ulrich[1]**

[1]Los Alamos National Laboratory, [2]National Nuclear Security Administration, [3]Pacific Northwest National Laboratory, [4]Texas A&M University, [5]Oak Ridge National Laboratory, [6]Uppsala University, [7]Lawrence Livermore National Laboratory, [8]Idaho State University, [9]University of Michigan, [10]Korea Atomic Energy Research Institute, [11]Swedish Nuclear Fuel and Waste Management Company, [12]Lawrence Berkeley National Laboratory, [13]University of California, Berkeley
E-mail: tobin@lanl.gov

***Abstract:***

*The Next Generation Safeguards Initiative Spent Fuel (NGSI-SF) Project started in 2009 with the general purpose of strengthening the technical toolkit of safeguard inspectors and the focused goal of measuring the Pu mass in spent fuel assemblies. Subsequently the safeguards goals of the projects have evolved to include verifying the correctness and completeness of the declaration. The first two years of the NGSI-SF Project research primarily involved Monte Carlo simulations used to quantify the expected non-destructive assay (NDA) signals from 14 different NDA techniques. In 2011, advised by an external review committee, the research focus evolved to the fabrication and fielding of a down-selected set of instruments as well as quantifying the expected performance of integrated systems. In 2013, we will start deploying NDA systems in collaboration with colleagues in Japan, the Republic of Korea and Sweden. This paper will describe the development of prototypes and the planned field trials of each deployed system. Additionally the general evolution in the overall NGSI-SF research project will be described including description of, and motivation for, the six NGSI-SF assembly libraries, and noteworthy technical highlights.*

**Keywords:** non-destructive assay; spent fuel; used fuel; Pu mass; heat; declaration completeness; declaration correctness

## 1. Introduction

In 2009, the Next Generation Safeguards Initiative Spent Fuel (NGSI-SF) Project of the U.S. Department of Energy's National Nuclear Security Administration (DOE/NNSA) began a nominally 5-year effort to research the capability of an integrated nondestructive assay (NDA) system to improve the state-of-the-practice in safeguards technology.[1,2] The technical safeguards goals of the NGSI-SF Project currently include the traditional NDA goals of verifying the "correctness" (i.e. initial enrichment [IE], burnup [BU], and cooling time [CT]) and the "completeness" (i.e. pins have not been removed) of a States spent fuel assembly declaration and the non-traditional safeguards goal of measuring the Pu mass in an individual assembly. Furthermore, the goal of accurately estimating the heat emitted from an assembly has also been added to the research plan due to the NGSI-SF Projects

collaborative work with the Swedish Nuclear Fuel and Waste Management Company (SKB) and Uppsala University. Although determining the heat of individual assemblies is not a safeguards requirement, combining this need of the Swedish Central Storage of Spent Nuclear Fuel and Encapsulation Plant (CLINK) with the safeguards needs listed above provides the potential to combine resources, and hopefully, produce a more effective and efficient system among the multiple needs.

## 2. Current State-of-the-Practice in Spent Fuel Nondestructive Assay

Before describing how the NGSI-SF Project hopes to improve upon the state-of-the-practice, a brief description of the status quo is in order. The primary NDA instruments used by inspectorates world-wide today can be divided into three general groups: (1) Cerenkov viewing devices (ICVD or DCVD)[4,5] for which measurements are performed from the bridge crane, (2) detectors that measure the presence of $^{137}$Cs radiation (SFAT or IRAT) by lowering a small detector into the storage pool and (3) integrated total neutron and passive gamma measurement systems (FDET or SMOPY)[6,7,8] that are positioned in a well-controlled manner relative to the fuel in the pool.

The capabilities of these instruments have shaped the policies of inspectorates such as the International Atomic Energy Agency (IAEA). Because measuring Pu mass in assemblies is such a challenging endeavour, measurement based Pu accountancy is not currently done with spent fuel assemblies. Instead measurement based Pu accountancy starts after the Pu mass is removed from the fuel. Before reprocessing the protection and safeguarding of Pu depends largely on physical protection, calculated Pu mass accountancy, which depends on unverified facility data, and maintaining continuity-of-knowledge (COK). NDA measurements of spent fuel have primarily focused on satisfying the IAEA requirements of the "gross defect" test, for which one is interested in detecting attributes consistent with spent fuel (i.e. there is a spectral line from $^{137}$Cs); and the "partial defect" test, for which the goal is determining that "*at least half of the fuel pins are present*" at the 90% or better confidence level.[9] With the current practices in mind, the NGSI-SF project was initiated with the intent of illustrating possible improvements to the status quo. The intention was also to inform policy makers and facility operators. The research aimed to describe how much better the results could be and what burden, if any, might be required by new technologies. Ideally some of the technologies will provide benefits to both the facility and the inspectorate.

In the context of the four NGSI-SF technical goals (correctness, completeness, Pu mass and heat), the following is a brief summary of the status quo: (1) The correctness of the declaration (IE, BU and CT) is not directly measured, rather the declaration, when researched, is most commonly checked for consistency by an integrated total neutron and passive gamma measurement systems. (2) The completeness of the declaration is most often checked by observing the Cerenkov glow of the water around the fuel pins or the presence of $^{137}$Cs spectra emitted from an assembly is checked. (3) Pu mass is generally not measured for individual assemblies. A "pass/fail" type measurement on fissile content of individual assemblies was performed at Thorp in order to primarily meet the needs of the facility.[10] And (4) in the context of accurate heat measurements, such measurement have already been made at the Central Storage of Spent Nuclear Fuel (CLAB) in Sweden with an assembly length calorimeter; additionally the use of passive gamma measurements for inferring heat have also been researched at CLAB.[11,12,13] The role for the NGSI-SF Project in this context is to investigate if the uncertainty in the heat determination can be reduced by using other/additional NDA signatures coupled with simulation capabilities.

## 3. Approach to Researching NDA Technologies for Spent Fuel

The measurement of spent fuel with NDA is complex due to the highly variable isotopic composition of spent fuel. What goes into a reactor as a rather simple isotopic entity consisting primarily of $^{16}$O, $^{235}$U, $^{238}$U; comes out of the reactor comprised of a very wide range of isotopes that can impact an NDA signal. Of particular note to gamma NDA instruments is the fact that the gamma emission is overwhelmingly dominated by fission fragment. Of particular note to neutron NDA instruments is the fact that the neutron emission is dependent primarily on the build-up of $^{244}$Cm (and a few other isotopes) and the multiplication of the $^{244}$Cm origin neutrons. This multiplication depends on the presence of fissile isotopes, fertile isotopes and neutron

absorbers. All this complexity helps and hurts in the application of NDA in pursuit of our 4 technical goals. Of particular note in the context of verifying the correctness of a declaration, the complex combination of the fissile isotopes, fertile isotopes and neutron absorbers appear in simulations to enable neutron techniques such a differential die-away to determined IE, BU and CT in novel ways that are penetrating of the entire assembly.

In order to research this complex fuel given the range of safeguards needs for both portable and fixed NDA installations, the NGSI-SF project approach included the following components: (1) A wide range of NDA techniques were studies, 14 individual measurement techniques were evaluated through simulation and limited experimentation.[1,2] (2) A range of spent fuel libraries were simulated that captures the isotopic complexity of fuel; 64 spent fuel assemblies are in the first library and a few hundred more have subsequently been added.[14,15] (3) An extended period of time was dedicated, 5 years being the initial estimate.[1,2] And (4) the integration among techniques was considered important from the start as is explain in the next section.

Efforts have now turned to the construction of 4 or 5 integrated systems comprised of the most promising and complementary techniques that are ready for near- to medium-term implementation. Field demonstrations with international partners are in process for the coming years starting in June of 2013 with Japan and with Sweden, and in the fall of 2013 with the Republic of Korea.

## 4. Motivation for an Integrated System of NDA Technique

The motivation for NDA technique integration can be thought of as being driven by the isotopic complexity. Each NDA signature constrains the range of possible content of the fuel. Each signature is sensitive to diversion or replacement of pins in a different way; hence, the integrated system should have a lower threshold for missing/diverted material than the status quo. As an example, in general all the NGSI-SF systems add at least one signature to the signatures measured by a Fork Detector (FDET per IAEA) or enhanced Fork Detector (FDET with CZT detector). The total gamma signal of FDET measures around 90% of its signal from the outer 2 rows of the assembly. This signal may be either spectral resolution or simply the total gamma signal as measured by an ion chamber. In general, particularly for longer cooled fuels, this signal is proportional to the BU of the edge of the fuel. The neutron signal is sensitive to the entire assembly because all commercial spent fuel assemblies experience significant neutron multiplication even if stored in borated water. This neutron signal, on the other hand, is particularly sensitive to the build-up of $^{244}$Cm but also to the depletion of fissile material and the build-up of neutron absorbers. To these gamma and neutron signals, the integrated NGSI-SF systems add additional signatures that further constrain the composition of the fuel; note: constraining the isotopic content of the fuel to a narrower range is equivalent to improving our ability to determine the correctness and completeness or the declaration, as well as quantifying Pu mass and heat.

Below the unique signatures of the added NDA capability of the different NGSI-SF techniques are listed:

1. Passive Neutron Albedo Reactivity[17,18,19,20] (PNAR) effectively measures the multiplication of thermal neutrons incident on all sides of the assembly.
2. Californium Interrogation Prompt Neutron[10,21] (CIPN) measures the multiplication of neutrons originating from a $^{252}$Cf source on one side of the assembly as detected on the opposite side of the assembly.
3. Self-Interrogation Neutron Resonance Densitometry[22,23,24,25] (SINRD) measures the perturbation of the neutron energy spectrum by the outer few rows of pins at the $^{239}$Pu resonant energy of ~0.3 eV.
4. Differential Die-Away Self-Interrogation[26,27] (DDSI) measures doubles and singles count rates from which multiplication can be determined. The full extent to which correlated neutrons in various time windows can be exploited to achieve our 4 goals is a topic of active investigation.
5. Differential Die-Away[28,29,30] (DDA) measure the neutron intensity and neutron die-away time constant during several window in time following a neutron burst originating in a neutron generator on one side of the fuel.

## 5. Deployed Systems

### 5.1 Passive Neutron Albedo Reactivity + Passive Gamma + Total Neutron:

The PNAR detector being deployed at the Fugen Reactor site in Japan is depicted in Figure 1. This detector has a “high” multiplying section created by surrounding the fuel with polyethylene. It also has a “low” multiplying section created by lining the inner surface of the detector with Cd; Cd is used because it absorbs effectively all neutrons incident upon it with energies below ~0.5 eV. By taking the ratios of the count rates from fission chambers located in each of these sections, the PNAR signature or “Cd ratio” is calculated. The Cd ratio is proportional to the multiplication of the fuel which is a function of both the fissile content (the weighted sum of $^{235}$U, $^{239}$Pu, and $^{241}$Pu) and the neutron absorber content.

The detector depicted in Figure 1 also has additional fission chambers similar to those in a SINRD detector. They will be used to benchmark the relative intensity of different parts of the neutron energy spectrum. Due to the circular design and the large water gap (7.5 mm on all sides) between the fuel and detector wall, it is not expected that a SINRD analysis will be possible.

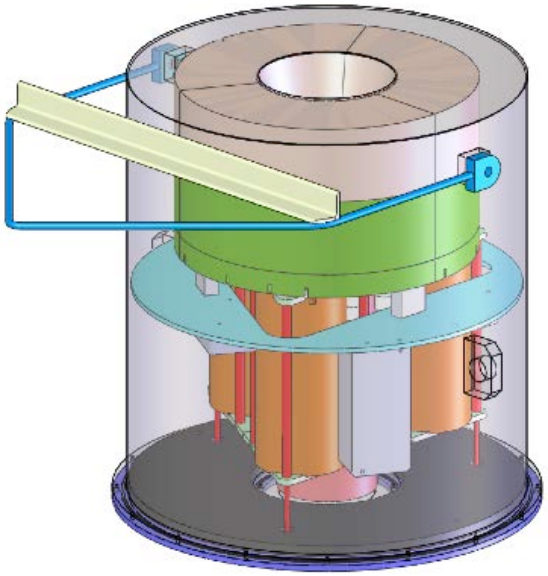

**Figure 1:** Passive Neutron Reactivity Detector designed for the measurement of Fugen fuel.[31]

The current plan is for 8 assemblies to be measured at Fugen in June of 2013. Seven of the eight assemblies are MOX assemblies that vary in BU (3 to 19 GWd/T) and CT (10, 15, 20 yrs); one assembly is LEU fuel (1.9% IE, 7 GWd/T BU, 10 yr CT). The Fugen fuel will provide a good test of the sensitivity of PNAR because the fissile content in the fuel does not vary much even as the BU does. This is because the fuel started with rather low fissile loading because Fugen is a heavy water moderated reactor with a small physical cross sectional area (28 rods). Additionally, the water gap between the fuel and the detector wall matches that of the Fugen storage racks; this water gap is double that of a traditional commercial reactor (3.2 mm on all sides). This is noteworthy because the larger the water gap, the lower the impact of the Cd liner on the multiplication of the fuel. Following deployment at Fugen, the PNAR detector will be returned to LANL where measurements will be performed with fresh PWR fuel rods over a range of average IE values (0.2%, 0.7%. 3.2%, 3.2% with Gd) for two different rack sizes.

### 5.2 Californium Interrogation Prompt Neutron + Passive Gamma + Total Neutron:

As illustrated in Figure 2, the CIPN instrument is nearly identical to that of a Fork detector. The CIPN signal is calculated by subtracting the count rates from two separate measurements. The first measurement is the passive neutron background count rate. For the second measurement, a $^{252}$Cf source is brought in close to the assembly; to a point that is ~5 cm from one side of the assembly. The $^{252}$Cf source is positioned on the opposite side of the assembly from the fission chamber detectors. This second measurement quantifies the combined count rate of the passive neutron background and the neutrons that were produced by the multiplicative chain reaction that started with $^{252}$Cf source neutrons. Given the size of the assembly and the dimensions of the detector, the probability of a $^{252}$Cf source being directly detected in the fission chamber is small. The net signal above the background is overwhelmingly due to the fission chain reaction that multiplies across the assembly and as such is proportional to the multiplication in the assembly and is sensitive to fuel pins in all parts of the assembly.

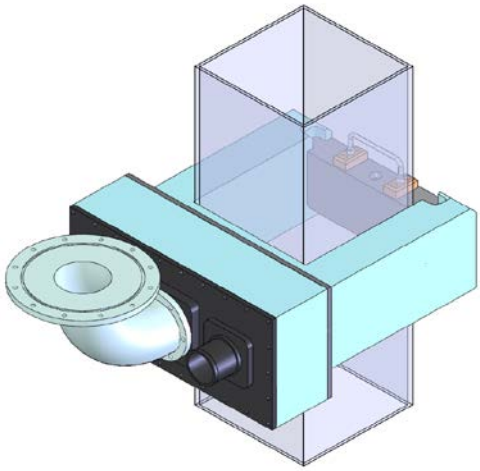

**Figure 2:** Californium Interrogation Prompt Neutron detector designed for the measurement of PWR fuel at the Post Irradiation Examination Facility in the ROK.[31]

The current deployment plan is to measure 6 or 7 assemblies in the fall of 2013 at the Post Irradiation Examination Facility (PIEF) located in Daejeon, ROK. All the assemblies are LEU, PWR assemblies that vary in BU from 17 to 38 GWd/T, CT from 21 to 34 years, and IE from 2.1% to 3.5%. All but one of the assemblies is a 14 x 14 and all but one of the assemblies is contained within metal shrouds. The shrouds are necessary in order to move the assemblies around because the assembly tops were cut off to remove a few pins for post irradiation examination. Because the shroud is fabricated of metal, it is not expected to impact the CIPN measurement significantly. Before deployment at PIEF, the CIPN detector will be used to measure fresh PWR fuel rods of variable average enrichment (0.2%, 0.7%. 3.2%, 3.2% with Gd) in a 15 x 15 rack in both fresh and borated water.

### 5.3 Self-Interrogation Neutron Resonance Densitometry + Passive Gamma + Total Neutron:

The SINRD detector being deployed in ROK measures the neutron intensity in four different parts of the neutron energy spectrum using $^{235}$U fission chambers. The different energy sensitivities are obtained by surrounding the fission chambers by different absorbing, or "filtering," materials (cadmium, gadolinium, hafnium, and boron); or in the case of one tube, leaving it totally without any filtering material.

The energy spectrum that exists in spent fuel is driven by the birth of neutron, due to spontaneous or induced fission, with a Watt energy spectrum peaked near 2 MeV. These neutrons slow down, colliding multiple times in both the fuel and the water, with some neutrons making it to thermal energies. Because of these collisions, the neutron population at any given energy is influenced by the absorption cross section of the materials in the fuel and the water. The strong absorptive resonance (~3,000 b peak) of $^{239}$Pu at 0.3 eV is of particular interest to SINRD because the count rate difference between a gadolinium lined fission chamber and a cadmium lined fission chamber is particularly sensitive to neutrons in the 0.3 eV part of the spectrum. Hence, if there is a significant amount of $^{239}$Pu in the fuel, then there will be relatively few neutrons leaving the fuel with an energy of 0.3 eV. It is expected that a SINRD instrument can be sensitive to this absence of neutrons with and energy of 0.3 eV; with proper calibration and careful implementation, the expectation is that the mass to $^{239}$Pu can be measured for the outer few rows of one side of the assembly. Two important caveats are that the water gap between the fuel and the detector needs to be eliminated (or possible corrected for); and, for one cycle fuel, a correction for the impact of $^{235}$U is likely needed when using $^{235}$U fission chambers.[25] Note that the use of $^{239}$Pu fission chambers is expected to noticeable improve the sensitivity of SINRD to $^{239}$Pu; such research is being performed at SCK CEN in Mol Belgium

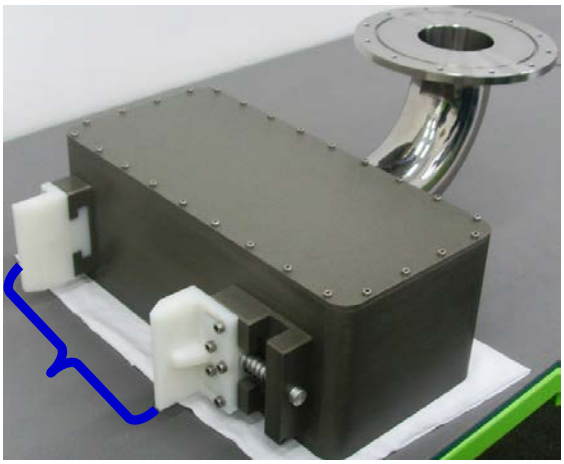

**Figure 3:** Self-Interrogation Neutron Resonance Densitometry detector designed for the measurement of PWR fuel at the Post Irradiation Examination Facility in the ROK.[32]The location of the fuel is indicated with blue bracket.

The current deployment plan is to measure 2 assemblies in the fall of 2013 at the PIEF in Daejeon, ROK. The small numbers of assemblies is driven by the fact that only one assembly (3.2% IE, 28 GWd/T, 28 yr CT, 14 x

14) at PIEF is not in a shroud. The shroud presents a particular concern for SINRD because the water between the fuel and the shroud on one side of the assembly can vary in a nonvisible way. A gap of between 0 and ~1 cm may exist. This magnitude of a water gap is expected to have significant impact on SINRD signal. Perhaps a correction for this water gap can be detected from the neutron energy spectrum. For this reason one assembly (3.2% IE, 38 GWd/T, 24 yr CT, 14 x 14) in a shroud will be measured as well. Before deployment at PIEF, the SINRD detector will be used at LANL where fresh PWR fuel rods (0.2%, 0.7%. 3.2%, 3.2% with Gd) will be measured in a 15 x 15 rack in both fresh and borated water.

### 5.4 Differential Die-Away + Passive Gamma + Total Neutron (+ others):

A DDA measurement begins with the burst of neutrons, ~1 million neutrons; so commercial, non-actively cooled, neutron generators are expected to be used. A 10 μs burst was used for the NGSI research that was produced by a DT neutron generator. Some of those burst neutrons slow down to near thermal energies (0.025 eV). Because the neutron detectors, fission chamber or very small $^{3}$He tubes, used with DDA are cadmium-covered only neutrons above ~0.5 eV can enter the detectors. After ~200 μs the burst neutrons account for ~1% of the net detected signal above the background; ~99% of the net signal comes from neutrons created by induced fissions that occur after the burst. Traditional DDA functions by count neutrons during a time window when the burst neutrons are negligible, so after ~200 μs, compared with the induced fission count rate. In recent publications[28,29,30] it was demonstrated in simulation that interesting information can be obtained by measuring during the time interval when the signal is a mixture of both burst neutrons and induced fission origin neutrons. Both types of neutrons provide information about the content of the fuel. In particular, the multiplication in all assemblies, spent and fresh, is correlated with the net total neutron count rate during the 100 to 200 μs time window. So there is the promise of absolutely measuring the multiplication in each assembly. Furthermore, simulations done with several of the NGSI spent fuel libraries[14,15] indicate that IE and BU, and CT to a lesser degree, can be determined from a graph of the total neutron count rate vs. the neutron die-away time in the 100 to 200 μs time window.

In Figure 4 a preliminary conceptual design of the DDA system planned for deployment in CLAB is depicted. This instrument will be deployed as part of a collaboration among SKB, Euratom, Uppsala University, and the U.S. Department of Energy (DOE). This collaboration is particularly exciting due to the wide range of fuel that will be measured as well as the quality and number of the NDA instruments that will be deployed. Because the CLAB facility has almost all the fuel used in Sweden over the past ~40 years on site, the IE, BU and CT varies considerable. It is anticipated that ~25 PWR and ~25 BWR assemblies, intentionally selected to span the range of IE and BU, will be measured. All these assemblies, becoming known as the "SKB-50," will be measured with the following NDA instruments:

1. DDA signal: total neutron count rate, neutron die-away time and total neutron intensity in different detector banks all for several different time windows.
2. Spectrally resolved gamma emission measured with an HPGe detector from all sides of an assembly.
3. Gross gamma emission measured with an ion chamber on two sides of the assembly.
4. Total neutron, which is essentially the background for the DDA instrument will be measured. It is worth noting this total neutron signal separately because its variation with BU (~BU$^{4}$) provides additional constraint on the isotopic content of the fuel.
5. Full assembly length calorimeter measurements.
6. Possible, correlated neutron detection with a DDSI instrument (more on this possibility in the next sub-section).
7. Likely, total neutron and gross gamma measurements from within the ~25 PWR assemblies obtained by inserting fission chamber and ion chamber down the guide tubes with the Partial Defect Tester instrument.[34,35,36]

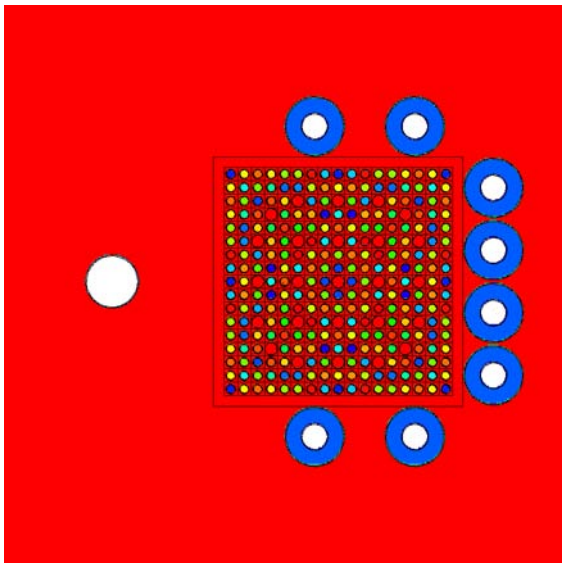

**Figure 4:** Conceptual Differential Die-Away detector designed for the measurement of PWR and BWR fuel (BWR insert not shown) at the Central Storage of Spent Nuclear Fuel facility in Sweden.

The measurement plan described above is accompanied by a simulation plan that involves modelling the fuel burnup with the Studsvik code and the SCALE code.[37] Note an insert is expected to be inserted inside the DDA instrument when measuring BWR fuel; the insert will likely only displace water so it will be comprised of air and metal. Before deployment at CLAB, the DDA detector will be used at LANL where fresh PWR fuel rods (0.2%, 0.7%. 3.2%, 3.2% with Gd) will be measured in a 15 x 15 rack.

### 5.5 Differential Die-Away Self-Interrogation + Passive Gamma + Total Neutron:

DDSI has "differential die-away" in its name because DDSI also has a burst; however, in the case of DDSI, the burst is a spontaneous or induced fission event that liberates nubar neutrons, so 1 to ~10 neutrons, 2 to 3 most commonly. DDSI has two signals of interest in the context of spent fuel. One signal uses the ratio of the doubles counts in an early time gate to the doubles counts in a late time gate. The other signal uses the ratio of the doubles count rate in the late gate to the singles count rate (D/S). The D/S ratio is a standard quantity used in classical coincidence counting to measure multiplication. What makes the DDSI doubles calculation unique is the use of a very long delay between the measurement of a neutron "trigger" and the opening of the gate. With DDSI the gate is delayed for the purpose of separating the passive interrogating signal, composed significantly of $^{244}$Cm, from a signal that overwhelmingly originates from induced fission.

Figure 5 illustrates the $^{3}$He–based DDSI instrument designed for deployment to either the PIEF in ROK or CLAB in Sweden. If it is deployed to the ROK it will likely measure ~4 assemblies. The number is slightly reduced from those CIPN will measure at PIEF because a different pool will be used to which less fuel can move. If deployed at CLAB, DDSI will measure the SKB-50. As will DDA, an insert may be used to enable more accurate measurements of BWR assemblies. One of the strength of DDSI, relative to other NDA techniques is that it is ~ 5 times less sensitivity to assembly positioning inside the detector;[16] research needs to be performed to the need for an insert. An additional strength of DDSI is that DDSI is totally passive. Furthermore, because DDSI and DDA share similar physics, it is hoped, although not well researched, that DDSI will share some of DDAs promise for measuring IE, BU and CT. Hence, there is some hope that DDSI will be useful for verifying the correctness of a declaration. Before deployment at PIEF or CLAB, the DDA detector will be used at LANL where fresh PWR fuel rods (0.2%, 0.7%. 3.2%, 3.2% with Gd) will be measured in a 15 x 15 rack.

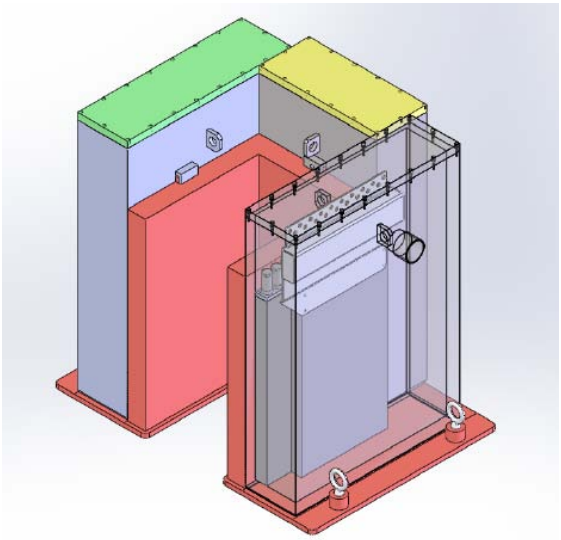

**Figure 5:** Differential Die-Away Self-Interrogation detector designed for the measurement of PWR fuel either the Post Irradiation Examination Facility in the ROK or the Central Storage of Spent Nuclear Fuel (CLAB) facility in Sweden.[31]

## 5. Future Plans and Summary

The NGSI-SF Project is in its final phase of NDA system deployment. The research effort

have been very fortunate to form collaborative relationships with researcher in Japan, the Republic of Korea and Sweden that are enabling the testing of the NDA techniques on spent fuel. Measurements are beginning with all partners in 2013. Some experimental plans are expected to extend into 2014 and 2015. The major focus of the research after experimental results are obtained, both with spent fuel in the field and LEU fuel at LANL, will be combining the various NDA measured results and simulations results to provide a defensible uncertainty estimate for the four quantities of interest: correctness and completeness of the declaration, Pu mass and heat.

## 6. Acknowledgements

The authors would like to particularly thank the parties sponsoring this research: Japan Atomic Energy Agency, Korea Atomic Energy Research Institute, Swedish Nuclear Fuel and Waste Management Company and the Next Generation Safeguards Initiative, Office of Nonproliferation and International Security, National Nuclear Security Administration.

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# Enhanced Data Authentication System: Converting Requirements to a Functional Prototype

**Maikael Thomas, George Baldwin, Ross Hymel,**

Global Security Programs, Sandia National Laboratories
Albuquerque NM USA
mthomas@sandia.gov; gtbaldw@sandia.gov; rwhymel@sandia.gov

**Andreas Smejkal, Peter Schwalbach, Morgan Rue,**

Nuclear Safeguards Directorate
European Commission Directorate-General for Energy
Luxembourg
andreas.smejkal@ec.europa.eu; peter.schwalbach@ec.europa.eu,
morgan.rue@ec.europa.eu

**Luc Dechamp and João G.M. Gonçalves**

Institute for Transuranium Elements, European Commission Joint Research Centre
Ispra, Italy
luc.dechamp@jrc.ec.europa.eu; joao.goncalves@jrc.ec.europa.eu

***Abstract:***

The Enhanced Data Authentication System (EDAS) is a technical concept to securely "branch" measurement data from operator-owned instrumentation to a Safeguards inspectorate, while guaranteeing the integrity of the operator communication link. While Safeguards normally depend on measurements that are fully independent from those of the operator, certain situations may call for the sharing of information from facility systems for both operations and verification purposes. An inspector must be confident that this branched information is a secure, true, and complete replica of the operator instrumentation. At the same time, an operator must have the assurance that the branching does not introduce an unacceptable risk to facility operations.

The EDAS project is a joint collaboration between the European Commission Directorate-General for Energy, the Institute for Transuranium Elements of the European Commission Joint Research Centre, the U.S. Department of Energy, and Sandia National Laboratories. Recognizing the special and conflicting requirements of the inspector and the operator, we have broken EDAS development into two phases. An initial EDAS prototype, focused on inspector requirements, was tested in a laboratory setting using representative instrumentation based on serial (RS232) communication. Results of these tests show that EDAS is able to meet inspector requirements. Current development emphasizes the operator concerns: establishing the complete set of requirements, designing and implementing a solution, and testing performance.

In this paper we focus on the second phase. We have developed an improved functional prototype that incorporates operator requirements. Considering these requirements, we show how they motivated the selection of our chosen embedded platform and accompanying Linux operating system. For software, we used open source libraries for communications and to encrypt and authenticate the data. We plan an installation and field test of the EDAS prototype at a facility in the United Kingdom in late 2013 to demonstrate the EDAS concept.

**Keywords:** branching, measurement, operator, safeguards, authentication

## 1. Introduction

The process monitoring of safeguarded nuclear fuel cycle facilities has a history of technical and policy challenges. Typically, the instrumentation used for safeguards inspections and that used for facility operations are distinct, using separate inspector and operator equipment, which can reconcile the differing needs of the inspector and facility operator. However, there can be practical cost and space limitations to this approach. Due to these constraints, it could provide additional confidence to an inspector to supplement dedicated safeguards implementations with other measures that provide complementary information directly from operator instrumentation.

We present the Enhanced Data Authentication System (EDAS) as a minimally intrusive technology that could provide inspectors with complementary safeguards information and more comprehensive view of a facility [1,2]. The concept of EDAS is to tap, or "branch," an operator's existing instrumentation to provide the inspector a secure and identical copy of the information flowing to/from the sensor. A conceptual example of EDAS is illustrated in Figure 1. Examples of sensors could include temperature probe, a flow monitor, a weight scale, a switch, or any other instrument.

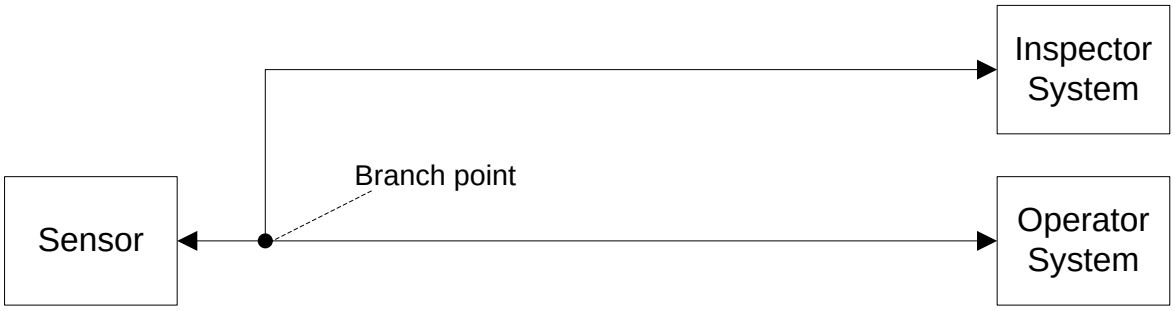


Figure 1: EDAS Branching Concept

For EDAS to be trusted and accepted by inspectors and operators, it is important to address the requirements of both parties. As part of a first phase in the development of EDAS, our team collected the inspector requirements and developed a first generation prototype that was demonstrated to members of the DG-Energy, the IAEA, and the Joint Research Centre. EDAS is now in a second phase of development that focuses on collecting operator requirements and creating a second generation prototype that combines both inspector and operator requirements. This paper focuses on converting these requirements into a functional prototype, and describing our architecture and design in greater detail.

## 2. Inspector and Operator Requirements

It is important to enumerate the inspector and operator requirements as it motivates the EDAS architecture and design.

### 2.1. Inspector Requirements

As listed in Baldwin [1], the requirements collected for inspectors are as follows:

**Accurate**
By accurate, we mean identical, on a bit-for-bit basis, to the information passed to the operator by the primary signal chain.

**Complete**
The inspector wants to see all of the information passed in the signal chain between the sensor and the operator. The signal chain may be bidirectional—with control signals being passed to the sensor—

so a branching solution would want to capture both data streams. If there should be multiple (parallel) data paths, each would need to be branched in duplicate.

**Authentic**
The inspector wants assurance that the information indeed comes from the agreed branch point without change—whether by unintentional interference, loss of integrity, or deliberate manipulation. For digital signals, this requirement generally calls for cryptographic signing of the branched data ("authentication").

**Meaningful**
At its most basic level, the branched data stream of 0's and 1's is meaningless out of context. An inspector needs to be able to interpret the data stream in the same way that the operator does. To use an exaggerated example: It does no good for the inspector to observe the messages "off" and "on" confidently, if the operator instead interprets these messages to instead mean their opposites, "on" and "off." The data stream is interpreted literally; the operator must not be talking in secret code.

**Confidential**
As a general rule, an inspectorate is expected to treat safeguards information as confidential. Third parties should not be able to eavesdrop. Thus the information branched to the inspector typically requires encryption.

## 2.2. Operator Requirements

The operator requirements were collected more recently as part of a second project cycle with EDAS. For an operator to feel comfortable allowing EDAS inside of a facility, our team assumes the following operator requirements:

**Noninterfering**
Once in place, an inspector branch cannot interfere in any way with the signal chain between the sensor and the rest of the operator's instrumentation. Data cannot be dropped, delayed, or altered. In every respect, the original signal chain must function exactly as it did without the inserted branch point. Essentially, noninterference is the single overriding requirement, but we mention two variants explicitly in the next two requirements:

**Fail-safe**
Noninterference must be the rule *whatever* the condition of the inspector signal branch. In particular, if the inspector branch should fail (e.g., EDAS loses power), the operator's signal chain must not be affected. Detailed requirements must anticipate and explore *all* conceivable modes of failure.

**Benign**
Noninterference also implies that an inspector could never *intentionally* manipulate the operator's sensor and instrumentation through the branch. The branch is strictly passive (unidirectional).

**Consistent with Instrumentation Standards**
Operator instrumentation may be compelled to comply with particular instrumentation standards, or satisfy specific performance criteria. For example, a measurement might function as part of an automated safety system. Instrumentation standards such as the Open Connectivity Standard (OPC) Unified Architecture are becoming more widely adopted. Any modifications for branching would need to satisfy such requirements, but these could vary depending on the particular measurement and on the facility.

**Provided with Bypass Option**
The operator must be able to physically bypass the inspector branch point if and when desired, for any reason, thereby leaving no question that the operator system had been restored fully to its original uninterrupted state.

## 3. System Architecture

The EDAS architecture can be divided into hardware/electronics and software components. By dividing the EDAS in this fashion, it will be easier to illustrate how the architecture and design meet the inspector and operator requirements.

### 3.1. Electronics Architecture

The EDAS electronics architecture is illustrated in Figure 2. This diagram includes several important architectural features starting at the tap off point of the operator instrumentation line and continues to the processor board that runs the EDAS software. Each major component of the electronics addresses either a requirement or a feature to enhance EDAS robustness.

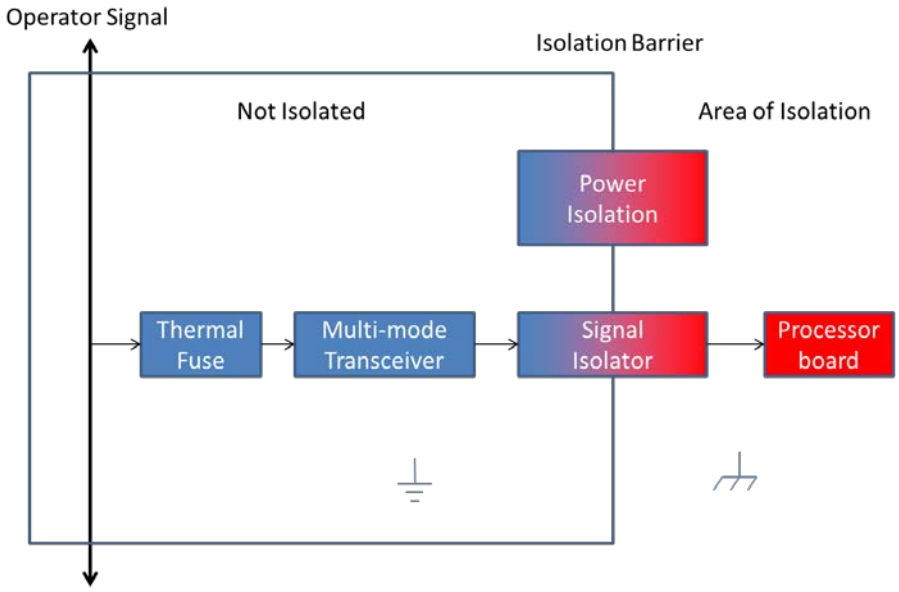


Figure 2: Electronics Architecture

The major electronics component that interfaces to the operator signal line is the thermal fuse. The thermal fuse addresses the noninterference operator requirement for fail-safe operations. In the case that the multi-mode transceiver or signal isolator electronics were to fail (e.g., due to overvoltage or incorrect polarity) and thereby interfere with the operator signal line, these fuses change their properties to become highly resistive, thus effectively isolating the signal line from the rest of EDAS. Similarly, if the EDAS electronics were to lose power, the electronics will act as an open circuit, preventing noise from reaching the operator signal. These steps help prevent potential corruption of data on the operator signal line in the case of multiple EDAS failure scenarios.

The multi-mode transceiver allows the processing of multiple types of signal inputs to make the EDAS more extensible. For example, such a transceiver could support RS-232 or RS-485 signals, so that the same EDAS could be deployed to tap instrumentation that uses either of these signal types.

Another important feature of the electronics is the various isolation features. All signals tapped from the operator signal line are passed through a signal isolator. An isolator ensures that EDAS is functionally benign as it allows for the operator signals to pass to the EDAS processor board while preventing any EDAS (and, by extension, inspector) signal from entering the operator signal line. A power isolator also provides isolated power for electronics inside of the EDAS to prevent power surges from destroying the components. Without a power isolator, in the worst case, a power surge could destroy the signal isolator, rendering it ineffective.

### 3.2. Software Architecture

The EDAS software architecture is shown in Figure 3. This diagram represents the major software blocks of the processor board block of Figure 2. As with the electronics, each major software component addresses a major inspector or operator requirement.

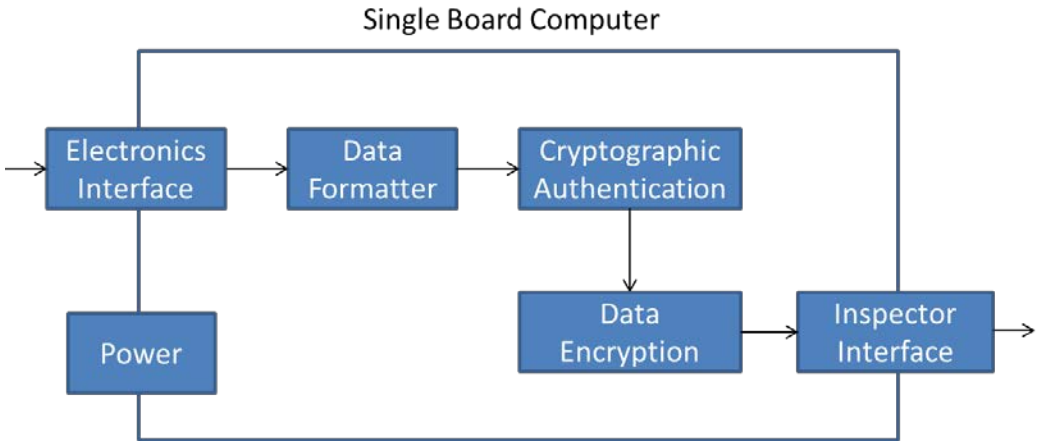


Figure 3: EDAS Software Architecture

Data from the operator signal line will first pass through a data formatter, whose function is to add a metadata tag to all incoming data packets. To simplify the EDAS architecture, the EDAS does not have an understanding of the underlying data passing through it. This task is left for subsequent processing by the inspectorate computer. However, to aid with downstream processing, the EDAS will add the following metadata tags: time stamp of first received packet, time stamp of last received packet, ID of input port, ID of EDAS, ID of tapped pin (e.g., Rx or Tx of RS-232 connection), and the number of bytes in the data block. The use of two timestamps implies variable length output messages from the EDAS, but this solves the issue where input data could be continuous or in bursts. The data formatter addresses the inspector requirements that the data be complete and meaningful as data packets will be formatted with helpful header information. The packets shall also be formatted to meet the operator requirement that data will be consistent with instrumentation standards. More information is given in the next section as to which specific standard our prototypes comply.

The EDAS will cryptographically authenticate all data packets to ensure that the inspector authentic requirement is met. The software will be configurable to allow a selection from various authentication algorithms that meet the stringent NSA Suite B standards for data security. After the authentication block, data packets are encrypted to the same NSA standards to meet the inspector requirement for data confidentiality. After these steps, all data packets are pushed to an inspector system, where the operator instrumentation data can be further analyzed.

## 4. Prototype Design

Our design of an EDAS prototype is influenced in part by an anticipated field trial deployment. We will support branching either four RS-232 pins or two RS-485 pins on the prototype to interface and tap data from the selected field trial equipment. All data are passed through a capacitive $SiO_2$ signal isolator rated to 6kV. We use a magnetic power isolator to provide power to the isolated electronics in the EDAS. These electronic components will be fabricated on a custom printed circuit board, which will interface to the software processor board.

As shown in Figure 4, we have selected the low cost, credit card-sized BeagleBone as the processor platform to run the EDAS software [3]. The BeagleBone is powered through the USB port and will push output data to the inspectorate computer via its Ethernet port. The EDAS prototype runs a variant of the Linux operating system on the BeagleBone computer.

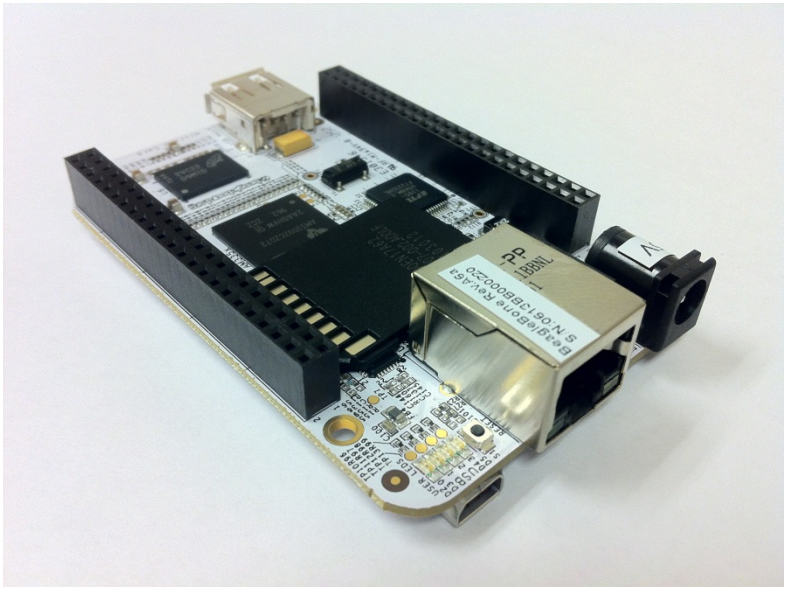

Figure 4: The BeagleBone Computer

All software is written in Java to support easy interface to available open-source libraries that support cryptography. As such, the EDAS prototype will support the selection of several NSA Suite-B cryptographic algorithms, such as the Advanced Encryption Standard (AES) for encryption and the public key Elliptic Curve Digital Signature Algorithm (ECDSA) for generating digital signatures.

To conform to data standards, the EDAS will output data that conforms to a format expected by the Remote Acquisition of Data and Review (RADAR) software package employed by the Euratom inspectorate for the automated acquisition of nuclear data [4]. The RADAR software will be hosted on an inspectorate computer connected to the EDAS via a network connection. As the EDAS pushes data to the inspectorate computer, a Data Acquisition Module (DAM), the entry point to RADAR, will automatically acquire, decrypt, and translate the data into a format understandable by the RADAR framework.

## 5. Testing and Field Trials

After completion of the EDAS prototype, we will perform a variety of tests to ensure it performs to functional requirements. The Joint Research Centre (JRC) in Italy will define and carry out specific tests consistent with the high-level test matrix detailed in Table 1. The tests by the JRC further will assure a European operator of acceptability for field trial deployment.

Table 1: EDAS Prototype: Test Matrix

| Test Set | Purpose |
|---|---|
| Branch correctness | Verify that the inspector and operator branches produce identical data. |
| Normal operations | Verify that the EDAS perform as required against several operational use cases. |
| Failure scenarios | Verify that the EDAS never interferes with the operator signal line based on various failure scenarios. |
| Tamper scenarios | Verify that the operator and inspector data paths cannot be manipulated by the other party. |

Once the JRC testing is has been satisfactorily completed, the Euratom safeguards inspectorate will work together with a facility in the United Kingdom to conduct a field trial of the EDAS prototype. We are planning an application that would share operator weight measurements of nuclear material cylinders. Two EDAS prototype units would be required. One branches the data from a weight scale; a second branches the data from an associated bar code reader used to identify the cylinders. The field trial will be an excellent opportunity to identify unexpected issues and better understand how EDAS could be most useful as a safeguards tool.

## 6. Conclusion

The EDAS is a secure branching concept that could provide complementary safeguards information to inspectors to create a more complete picture of facility activities. The design meets the requirements of both inspectors and facility operators. The upcoming testing and field trials of the EDAS prototypes will provide several opportunities to learn and further improve on the EDAS concept.

## 7. Acknowledgements

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# Application of $LaBr_3$ detector for neutron resonance densitometry

**H. Tsuchiya[1], H. Harada[1], M. Koizumi[1], F. Kitatani[1], J. Takamine[1], M. Kureta[1], H. Iimura[1], B. Becker[2], S. Kopecky[2], P. Schillebeeckx[2]**

[1]Japan Atomic Energy Agency (JAEA)
Tokai-mura, Naka-gun, Ibaraki 319-1195, Japan
E-mail: tsuchiya.harufumi@jaea.go.jp, harada.hideo@jaea.go.jp

[2]EC-JRC-IRMM
Unit D.4. Standards for Nuclear Safety, Security and Safeguards
Retieseweg 111, B-2440 Geel, Belgium
E-mail: peter.schillebeeckx@ec.europa.eu

***Abstract:***

*A method to determine the amount of nuclear materials in melted fuel resulting from a nuclear accident such as the one occurred at the Fukushima Daiichi nuclear power plants has not yet been established. The problem is complex due to the expected presence of $^{10}B$ and other strong neutron absorbing impurities. For this reason, neutron resonance densitometry, combining neutron resonance transmission analysis and neutron capture analysis, is proposed and a feasibility study has been defined. In this contribution a method to account for the presence of $^{10}B$ is presented and investigated. The study includes GEANT4 simulations to study the performance of a new well type gamma-ray detector based on $LaBr_3$ scintillators. In the design of the detector the main emphasis was on the capability to separate the full energy peak corresponding to the 478-keV gamma ray resulting from the $^{10}B(n,\alpha\gamma)$ reaction from the contribution of the 662-keV gamma ray due to the decay of $^{137}Cs$. In addition, experiments have been carried out at the time-of-flight facility GELINA of the EC-JRC-IRMM to test the capabilities of a $LaBr_3$ detector for NRCA applications, in particular to determine impurities present in the melted fuel. A neutron resonance capture gamma-ray experiment with a $^{nat}Se$ sample was performed using a $LaBr_3$ scintillator in parallel with a Ge-detector. The results of these measurements demonstrate that a $LaBr_3$ detector is suitable for NRCA as a part of neutron resonance densitometry.*

**Keywords:** neutron resonance capture analysis; $LaBr_3$ detector; neutron time-of-flight; melted fuel

## 1. Introduction

On 2011 Mar 11, a huge earthquake and subsequent gigantic tsunami caused a loss of electricity of the Fukushima Daiichi nuclear power plant and the system cooling the nuclear fuel installed at the reactors Units 1−3 stopped its operation. As a consequence, the nuclear fuel melted in the pressure vessel with even the risk that some melted fuel penetrated through the pressure vessel. Consequently, the possibility exists that melted fuel (MF) was deposited on the concrete floor and solidified together with other materials, resulting in a contamination of the buildings. At present, plans are made to remove the MF from the reactors after a cooling time of at least 10 years.

From the viewpoint of nuclear safeguards and security, special nuclear materials (SNMs) of uranium and plutonium present in the MF should be quantified at the time of after its removal. One of the possible techniques for such a

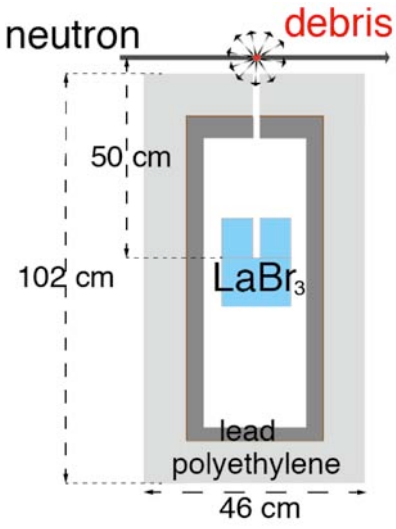


**Figure 1:** A schematic view of the $LaBr_3$ gamma-ray detector and its shields.

measurement is neutron resonance transmission analysis (NRTA). Bowman et al. [1] and Behrens et al. [2] successfully determined by NRTA the abundance of $^{239,240,242}$Pu and $^{235,236,238}$U in fresh and spent fuel pins with an accuracy better than 4%. Additionally, an extensive study based on Monte Carlo simulations was recently performed to investigate the feasibility of NRTA for the assay of plutonium in spent fuel from commercial light water reactors [3]. This study revealed the potential of NRTA to assay intact spent fuel assemblies. However, a method to measure SNMs in MF caused by a severe accident like the Fukushima case has not been established yet.

Unlike conventional commercial spent fuel, it is expected that the MF contains a substantial amount of impurities such as B, Si, Ni, and Fe. At present, no information is available about their relative abundance in the MF resulting from the Fukushima accident. In addition, particle-like debris of MF was formed due to e.g. a steam explosion [4]. Debris will also be produced during the process of removing the MF from the side. Such particle-like debris of MF will have a wide variety of size and composition, which complicates the measurements. Therefore, a method is needed which takes into account these difficulties. For this reason, neutron resonance densitometry (NRD) [5,6], which is based on NRTA combined with a kind of neutron resonance capture analysis (NRCA), is under development.

NRTA utilizes an intense pulsed white neutron source as interrogating radiation. The probability that a neutron beam is transmitted through the sample under investigation is studied as a function of neutron energy. The transmitted spectrum has characteristic dips. These dips are the result of the resonance structures in neutron induced reaction cross sections. The position and magnitude of these dips strongly depend on the content of the MF, i.e. on the relative amounts of the fission products and U- and Pu-isotopes present in the sample. Unfortunately, some of the impurities present in the MF have large total cross section for neutron interactions. As a consequence, their presence will result in a substantial reduction of the neutron flux and dilute the resonance profiles in the transmission spectrum. Among them, $^{10}$B present in both the control rods and the water that was poured for cooling has a large absorption cross section in the energy range of interest. It has been shown in Ref. [6] that the presence of $^{10}$B has an impact on the accuracy of the results obtained with NRTA. The accuracy can be improved when the amount of high absorbing impurities can be quantified. Since light elements like $^{10}$B do not have resonance in the low energy region, their relative abundance cannot be determined by NRTA or NRCA. Therefore, NRTA used for the analysis of SNM will be complemented by measurements of prompt gamma rays following neutron capture to determine impurities which do not have large resonances in the low energy region. Consequently, the full NRD system is based on NRTA and a combined use of NRCA and prompt gamma-ray activation analysis (PGAA). A description about NRD is given in Ref. [5], and in the contribution of Harada et al. [7] at this conference.

In this paper, the focus is on a combined use of NRCA and PGAA, referred to as neutron capture analysis, as a part of the NRD development. First, the role of neutron capture analysis in NRD is presented. Then, results of Monte Carlo simulations are discussed which demonstrate the capabilities of a well type $LaBr_3$ gamma-ray detector. This detector is especially developed to quantify the relative abundance of light elements such as $^{10}$B in the presence of a high background due to $^{137}$Cs. Finally, results of NRCA experiments performed with a cylindrical $LaBr_3$ at the neutron time-of-flight (TOF) facility GELINA of the EC-JRC-IRMM are presented.

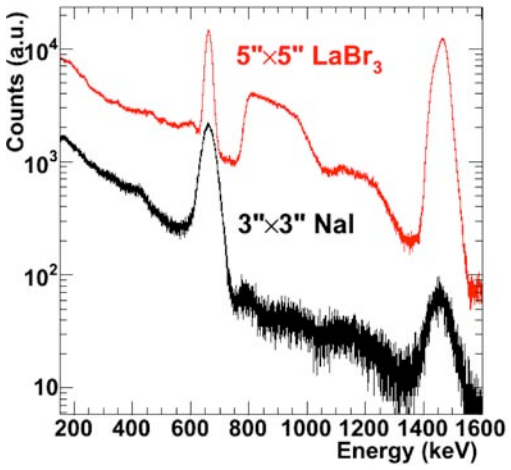


**Figure 2**: Energy spectra in 1000s, obtained by cylindrical $LaBr_3$ and NaI scintillators. The vertical axis indicates an arbitrary unit, because the distances from the $^{137}$Cs source to the two detectors are not the same.

## 2. Neutron Capture Analysis

### 2.1. Prompt gamma-ray emission and gamma-ray background in MF

Almost no information is available about the elemental and isotopic composition of MF after a severe nuclear accident. Very likely it will consists of a mixture of

SNMs, fission products, structural materials, steel and concrete. Thus, the presence of impurities such as B, Si, Cr, Ni, Fe, and Zr are to be expected. Neutron capture reactions in these impurities result in the emission of prompt gamma rays with energies ranging from a few hundred keV to about 10 MeV. In particular prompt gamma rays with an energy of 478 keV, 3534 keV, 7939 keV, and 1205 keV are emitted after neutron capture in $^{10}$B, $^{28}$Si, $^{52}$Cr, and $^{90}$Zr, respectively. Thus, an identification and quantification of these nuclides, and in particular of $^{10}$B, can be done by a prompt gamma-ray measurement.

Unfortunately, the detection of the prompt gamma rays and especially the 478 keV gamma ray resulting from the $^{10}$B(n,$\gamma\alpha$) reaction will be hampered by a high gamma-ray background due to the presence of radioactive fission products like $^{137}$Cs. The radioactivity of typical debris can be estimated from gamma-ray spectroscopic measurements of eight samples of debris originating from the Three Mile Island-2 accident [8]. More than 15 years after the accident, these debris samples are still radioactive mainly due to the presence of $^{137}$Cs (30.2 y), $^{134}$Cs (2.1 y), $^{154}$Eu (8.6 y), and $^{60}$Co (5.3 y). The number in parentheses indicates the half-life. From an analysis of the gamma-ray spectra the specific activity at the time of the accident were derived. The results are: $9.4\times10^{5}-3.4\times10^{8}$, $2.5\times10^{5}-8.5\times10^{7}$, $1.9\times10^{6}-2.9\times10^{6}$, and $7.8\times10^{4}-2.8\times10^{6}$ Bq/g, for $^{137}$Cs, $^{134}$Cs, $^{154}$Eu and $^{60}$Co, respectively. These values indicate that the main background affecting the background for neutron capture analysis is caused by the decay of $^{137}$Cs.

## 2.2. Neutron capture analysis detection system

To determine the quantity of impurities by neutron capture analysis in the presence of a high background due to$^{137}$Cs, a gamma-ray detector is required with a fast time response and a good energy resolution. The fast response is needed to avoid problems due to dead time, while the energy resolution is essential to separate in the pulse height spectra the prompt gamma rays resulting from neutron capture from the background due to the presence of radioactive material. Furthermore, the data acquisition system has to be optimized in order to process detector pulses in high count rate conditions. As data acquisition system the VME module V1720 of CAEN, which is a 8 channel 12 bit 250 MHz flash ADC waveform digitizer, is under investigation. For these studies measurements have been carried out with NaI and $LaBr_3$ scintillators at a neutron TOF facility of the Japan Proton Accelerator Research Complex (JPARC). The analysis of the data is in progress and the results will be published later.

### 2.2.1 Gamma-ray detector

A schematic representation of the gamma-ray detection system for neutron capture analysis is given in Figure 1. In the figure the location of the debris samples is also indicated. The gamma-ray detector (Fig. 1) has a well type shape to reduce the Compton background resulting from the 662-keV gamma rays emitted by $^{137}$Cs. This reduction is indispensable for the detection of the 478 keV gamma ray from neutron capture in $^{10}$B. The gamma-ray detector is made out of $LaBr_3$ scintillators. They are in the form of a tube or a cylinder, with a diameter of 12.7 cm. The height is 10 cm for the tube and 12.7 cm for the cylinder. The tube scintillator has a hole with a diameter of 2 cm. Each scintillator is packed in an Al container with a 0.5 mm wall

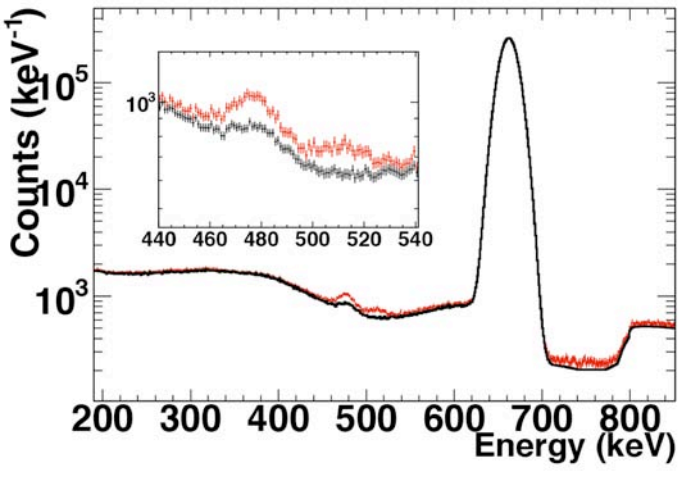


**Figure 3**: Pulse height spectra of the well type detector assuming 1h measurement time, calculated by the GEANT4 simulation. Red and black histograms correspond to an expected spectrum and background one, respectively. An inset shows the spectra at energies of $440-540$ keV. Errors are statistical ones.

thickness. To reduce the gamma ray and neutron background, the central gamma-ray detector is shielded with lead, silicon rubbers containing $B_4C$ (40 wt%), and polyethylene. They act as a collimator for the gamma rays emitted by the debris. The diameter of the polyethylene hole is 2 cm, while for the other shielding materials the diameter is 1.5 cm.

A $LaBr_3$ scintillator has a fast timing with a rise time of about 30 ns [9]. In addition, the energy resolution is better compared to the one of other inorganic scintillators. Figure 2 compares an experimental pulse height spectrum collected with a $LaBr_3$ scintillator (12.7 diameter and 12.7 cm height) with the one collected with a NaI scintillator (7.62 cm diameter and 7.62 cm height) in case of an isotropic irradiation by a $^{137}Cs$ source. The background due to environmental (e.g. $^{40}K$) and internal radioactivity ($^{138}La$ for the $LaBr_3$ scintillator) are not subtracted. From the spectra (Fig. 2) an energy resolution at 662 keV of 3.7% for the $LaBr_3$ and 7.4 % for the NaI scintillator was computed. The resolution of a Ge detector, about 0.5% at 662 keV, is much better. However, they cannot be used because their time response is not fast enough to perform measurement in a high gamma-ray background.

### 2.2.2 Results of Monte Carlo simulations

To verify the performance of the detection system (Fig.1), Monte Carlo simulations have been carried out using GEANT4. The detector response for a 478-keV gamma ray emitted from a debris sample was simulated in a 662 keV gamma ray background equivalent to a $^{137}Cs$ amount of $10^8$ Bq/g. This specific activity was derived from the TMI-2 debris measurements in Ref. [8]. The debris sample, with a mass of about 8 g, was contained in a cylinder with a 1 cm diameter and 1 cm height, corresponding to a density of 10.1 g/cm$^3$. The debris consisted of $^{nat}B$, $^{235}U$ and $^{238}U$ with a relative abundance of 10 wt%, 1wt% and 84wt%. The remaining of the sample was filled with 1wt% Pu and 4wt% O. This composition was based on typical spent fuel of a boiling water reactor with a burn-up of 40 GWd/t [10]. For a neutron capture analysis of the debris sample a neutron beam with an intensity of $10^{12}$ /s was assumed. The neutron energy spectrum was calculated with MCNP5. The results in Figure 3 reveal that the peak from the 478 keV gamma ray is superimposed on a high background resulting from the decay of $^{137}Cs$ and $^{138}La$. From an analysis of the spectrum one concludes that a 1h measurement is needed to obtain a net peak area of the 478 keV full energy peak with an 8% uncertainty only due to counting statistics.

## 3. Test experiment using a $LaBr_3$ detector at GELINA

To investigate the applicability of NRD, the Japan Atomic Energy Agency and the Institute for Reference Materials and Measurements of the Joint Research Centre (EC-JRC-IRMM) started collaboration in 2012. Within the collaboration various experiments are scheduled at the time-of-flight (TOF) facility GELINA of the EC-JRC-IRMM [11,12]. The experiments concentrate mainly on the impact of the characteristics of the debris on results of NRD, in particular the influence of the distribution of the particle size, the presence of neutron absorbing impurities, and the radioactivity of the sample.

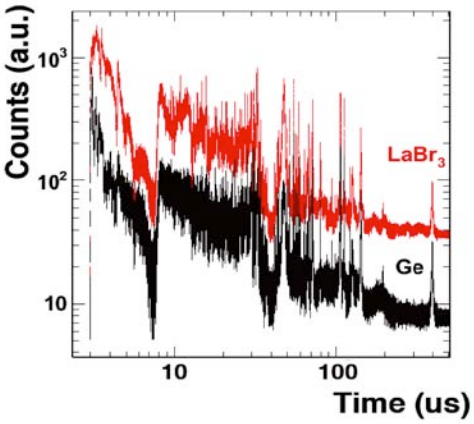


**Figure 4**: TOF spectra for the Se sample. Vertical axis represents an arbitrary unit. These are preliminary results.

Within this collaboration the performance of a $LaBr_3$ scintillation detector for neutron capture analysis applications was studied at GELINA. At this facility neutrons are produced via a photonuclear reaction by accelerated electrons impinging on a rotating neutron target consisting of a U-Mo alloy. After being moderated by light water, neutrons travel through vacuum flight paths to measurement stations in which detectors and samples are arranged. There are 10 flight paths with a length ranging from 10 m up to 400 m. Several measurement stations are installed at various nominal distances of 10, 30, 50, 60, 100, 200, 300, and 400 m. A detailed description of this TOF-facility can be found in Ref. [13].

The experiments to test the $LaBr_3$ scintillator were carried out at a 30 m flight path station. They started in November 2012 and finished in February 2013. Together with the $LaBr_3$ and a Ge detector was used. The $LaBr_3$ and Ge detectors were placed at 125° and 150° with respect to the incoming neutron beam, respectively. The $LaBr_3$ detector was cylindrical in shape with a 7.62 cm diameter and a length of 7.62 cm. The Ge detector was a coaxial detector with a 7.75 cm diameter and 7.65 cm length. To calibrate both detectors in energy and to verify their energy resolution measurements with radioactive sources of $^{137}Cs$, $^{60}Co$, Th, and a Pu-C mixture were performed.

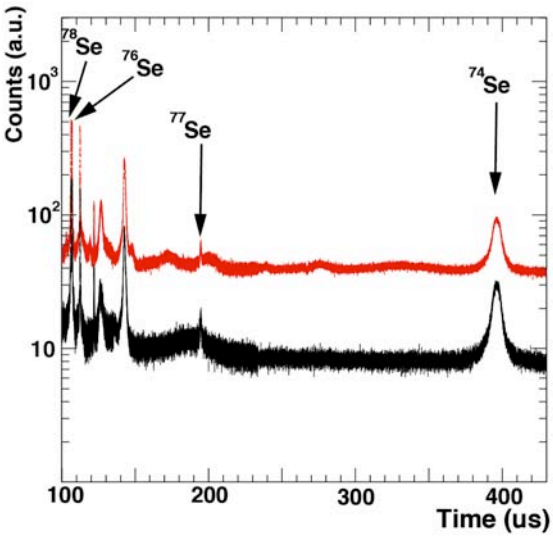


**Figure 5:** Same as Figure 4, but spectra between 100 – 430 us.

Four samples containing natural Se, S, Fe, and $B_4C$ were measured. The samples were placed at a 29 m distance from the neutron target, and located at a 13.5 cm and 25 cm distance from the $LaBr_3$ and Ge detector, respectively. The diameter of the beam at the sample position was 75 mm. In Table 1 a summary of the experiments is given together with some characteristics of the samples. The results of the measurements with the sulfur sample are used to correct the TOF-spectra taken with the Fe and Se sample for background due to neutron scattered by the sample. The measurements with the natural Fe and Se sample were carried out to verify the performance of a $LaBr_3$ detector for NRCA, i.e. using the resonance structures in the TOF spectra for elemental analysis. The measurements with the $B_4C$ sample were performed to verify the use of a $LaBr_3$ for the determination of the amount of $^{10}B$ by detecting the 478-keV gamma ray emitted in the $^{10}B(n,\gamma\alpha)$ reaction. A part of these measurements were carried out with a $^{137}Cs$ source placed in front of the $LaBr_3$ detector.

The analysis of the data is still in progress. In this paper, the potential of using a $LaBr_3$ detector for NRCA is demonstrated. For the Se sample the TOF spectra obtained with the $LaBr_3$ (red) and Ge (black) detector are compared in Figure 4. Even if the spectra are not yet corrected for the various background contributions, they exhibit the same resonance structures. The dips that appear in the TOF-region around 7 – 8 µs and 30 – 40 µs are due to the presence of the S and Na black resonance filters in the beam. Both spectra show also the same resonance structures resulting from neutron capture in the Se sample. This is emphasized in Figure 5 where the spectra are compared for TOF-values between 100 µs and 430 µs, corresponding to 430 eV and 23 eV, respectively. The TOF-spectra obtained with both the $LaBr_3$ (red) and Ge (black) detector clearly reveal the low energy resonances of $^{74}Se$ at 27.1 eV, $^{77}Se$ at 112 eV, $^{76}Se$ at 378 eV and $^{78}Se$ at 383 eV, corresponding to a TOF of 395 µs, 195 µs, 107 µs and 106 µs, respectively.

| Sample | Thickness (mm) | Diameter (mm) | Mass (g) | Background filters |
|---|---|---|---|---|
| Se | 8.0 | 80 | 66 | $^{10}$B, Na, S |
| S | 5.3 | 80 | 27 | $^{10}$B, Na, S |
| Fe | 1.0 | 80 | 40 | $^{10}$B, Na, S, Ag, W |
| $B_4C$ | 2.3 | 70 | 21 | $^{10}$B, Na, S, Ag, W |

**Table 1:** Summary of the test NRCA experiments.

## 4. Summary

In order to quantify SNMs in MF caused by the Fukushima accident, a method called Neutron Resonance Densitometry is under development. NRD combines neutron resonance transmission analysis and neutron capture analysis. One of activities within the R&D programme of NRD concerns the determination of the amount of $^{10}B$ to account for the flux attenuation in NRTA spectra used to quantify the amount of SNMs. In this paper a method relying on the detection the 478-keV gamma-rays produced in the $^{10}B(n,\alpha\gamma)^7Li$ reaction is proposed. To optimize the detection system for

measurements in a high gamma-ray background a well type gamma-ray detector consisting of $LaBr_3$ scintillators was designed. The performance of the well type detector was studied by results of Monte Carlo simulations. In the calculations the contribution of the background due to the radioactivity of $^{137}Cs$ was calculated using characteristics of debris samples originating from the TMI-2 nuclear accident. It was shown that a 1h measurement is needed to deduce the amount of $^{10}B$ from the net peak area of the 478-keV gamma ray with a 8% uncertainty resulting from only counting statistics. In addition to this study, time-of-flight experiments were carried out at the GELINA facility of the EC-JRC-IRMM as part of collaboration between the JAEA and EC-JRC-IRMM. The capabilities of a cylindrical 3"×3" $LaBr_3$ detector for NRCA applications were demonstrated by a comparing its performance with the results of TOF-measurements using a Ge-detector.

## 5. Acknowledgements

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# Recent modelling studies for analysing the partial-defect detection capability of the Digital Cherenkov Viewing Device

**Sophie Grape[1], Staffan Jacobsson Svärd[1]**

[1]Department of Physics and Astronomy, Uppsala University, Box 516, SE-75120 Uppsala, Sweden

***Abstract:***

*Strong sources of radioactivity, such as spent nuclear fuel stored in water pools, give rise to Cherenkov light. This light originates from particles, in this case electrons released from gamma/ray interactions, which travel faster than the speed of light in the water. In nuclear safeguards, detection of the Cherenkov light intensity is used as a means for verifying gross and partial defect of irradiated fuel assemblies in wet storage.*

*For spent nuclear fuel, the magnitude of the Cherenkov light emission depends on the initial fuel enrichment (IE), the power history (in particular the total fuel burnup (BU)) and the cooling time (CT). This paper presents recent results on the expected Cherenkov light emission intensity obtained from modelling a full 8x8 BWR fuel assembly with varying values of IE, BU and CT. These results are part of a larger effort to also investigate the Cherenkov light emission for fuels with varying irradiation history and other fuel geometries in order to increase the predicting capability and thus lower the detection limits for the Digital Cherenkov Viewing Device (DCVD).*

*The results show that there is a strong dependence of the Cherenkov light intensity on BU and CT, in accordance with previous studies. However, the dependences demonstrated previously are not fully repeated; the current study indicates a less steep decrease of the intensity with increasing CT. Accordingly, it is suggested to perform dedicated experimental studies on fuel with different BU and CT to resolve the differences and to enhance future predictive capability. In addition to this, the dependence of the Cherenkov light intensity on the IE has been investigated. Also, the modelling of the Cherenkov light emission has been extended to CTs shorter than one year. The results indicate that high-accuracy predictions for short-cooled fuel may require more detailed information on the irradiation history.*

**Keywords:** *DCVD, Cherenkov light emission, partial defects, initial enrichment, burnup, cooling time, Geant4*

## 1. Introduction

The nuclear safeguards system relies on the verification of operator declared information as well as on the confirmation of absence of undeclared activities. As part of the former category, inspections are continuously carried out at nuclear facilities, where different instruments are used to draw conclusions about the inventory of spent nuclear fuel. The Digital Cherenkov Viewing Device (DCVD) is one of the instruments that are available to inspectors at such occasions. Its principle is based on the detection of Cherenkov light; a type of light that is created in e.g. water around strong radioactive sources due to the release of highly energetic electrons, which move faster than the speed of light in water. These electrons are themselves products of the photoelectric effect and Compton scattering, caused by the interaction of gamma rays from decaying fission products with electrons in the water. Specifically for electrons moving in water (with a refractive index of 1.34), one can calculate that the electrons need at least a kinetic energy

of 257 keV in order to produce Cherenkov light. Larger electron energies lead to increased Cherenkov emissions, according to the Tamm-Frank equation [1].

For gross verification purposes, where inspectors draw conclusions on whether or not an item in a storage pool is a nuclear fuel assembly or a non-fuel item, the specific intensity of the Cherenkov light is not as important as its mere presence. However, for partial defect verification, where the goal is to determine if a fraction of the fuel rods in an assembly are removed or replaced, it is important to relate the detected Cherenkov light intensity to the intensity that is expected from that particular fuel assembly.

As a means to support partial-defect analysis, simulations of the expected Cherenkov light intensity as a function of burnup and cooling time are available [2] and have so-far been used for evaluations. These expected intensities are given in relative terms, and a calibration light source is needed for the absolute scale. However, some reasons have been put forward for repeating this type of simulations: (1) reported disagreements between simulated and experimental data, especially for cooling times below 5 years where large variations between the detected intensities from fuel assemblies with almost identical declared information but varying fuel irradiation history have been seen, (2) previous data do not include simulations for enrichments higher than around 2%, (3) previous data do not include simulations with uncertainty estimates, and (4) current simulation tools and computer power would enable more detailed simulations than previously possible.

By repeating and extending the previously performed simulations, the goal of this work is to obtain a measure of the modelling accuracy. In the longer term, we envisage improved capability to predict the Cherenkov light intensity from intact fuel assemblies, which will benefit the partial defect detection process and possibly lower the detection level of the DCVD to partial defects. The purpose of this paper is to report on new simulation results, obtained using state-of-the-art simulation codes for the same type of fuel as in the previous simulations, in order to allow for comparisons. An additional objective is to present results on the dependence of the Cherenkov light intensity on initial enrichment, which has previously not been reported.

## 2. Simulation software

A dedicated simulation tool for modelling the response in the Digital Cherenkov Viewing Device (DCVD) has been developed [3]. It comprises three main steps: 1) the generation of gamma-ray source terms in the fuel, 2) the modelling of the gamma-ray interaction (including the release of electrons in the water), particle transport and consecutive Cherenkov light emission, and 3) the transport of Cherenkov light from the fuel to the DCVD, including its detection in the device. In this work, only the Cherenkov light emission has been under study, and accordingly the last step has been omitted.

In the first step, the fuel's initial enrichment (IE), irradiation history and cooling time (CT) are taken into account, which is important since these parameters strongly affect the gamma-ray emission from the fuel, and thus the emission of Cherenkov light around the fuel. As an output, a gamma-ray spectrum is produced. In the earlier simulations, the source term generation was handled by the burnup calculation ORIGEN 2, while now ORIGEN-ARP [4] in the Scale6.1-package [5] is used. The new code includes more fuel types, a larger number of energy groups, and above all updated cross sections for neutron reactions based on continuous-energy and multi-group neutron data and coupled neutron-gamma data based on ENDF/B-VI.8 and ENDF/B-VII.0. Furthermore, the data libraries include decay data, neutron-induced fission product yields, delayed gamma-ray emission data, and neutron emission data based on ENDF/B-VII and JEFF-3.0/A. Photon yield data libraries are based on the recent evaluations from Evaluated Nuclear Structure Data File (ENSDF).

The second step is modelled using the Geant4 (version 9.0 patch 02) simulation tool [6], as compared to Geant3.15, which was used for the older simulations. Geant4 is a modern and updated version of the former and is written in object oriented C++. It is adapted to large-scale, accurate and comprehensive simulations of particle detectors used in nuclear physics experiments, radiation physics, space science

and nuclear medicine. Geant4 includes, among other things, new, refined and updated physics models, several different interaction models adapted to a variety of energy regimes and interaction types and offers a much better computing performance as compared to previous versions [6].

## 3. Modelling specifications

For all gamma-ray source term generation simulations included in this paper, gamma-rays with energies in the region of 0.257-3 MeV are extracted from ORIGEN-ARP, and here a total of 42 non-equal energy groups (intervals) were selected to provide sufficient energy resolution of the resulting gamma-ray spectrum. An example of a resulting gamma-ray spectrum is shown in figure 1. In the old simulations, only six broad energy groups were included.

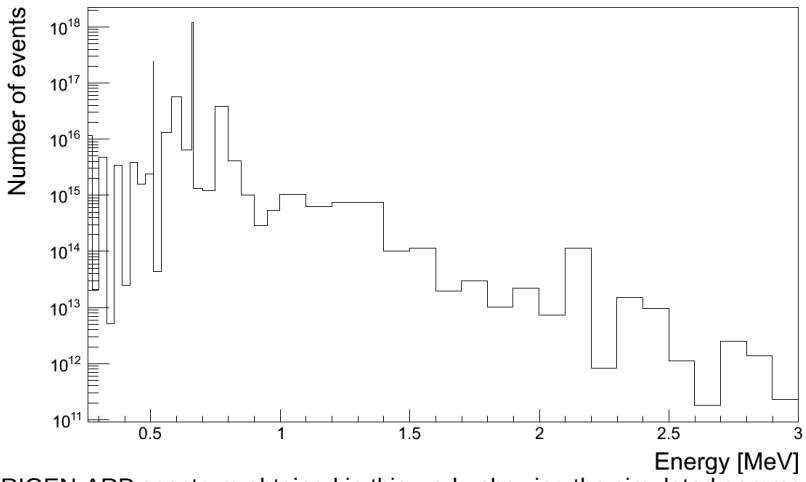


**Figure 1.** A typical ORIGEN-ARP spectrum obtained in this work, showing the simulated gamma-ray spectrum from an 8x8 fuel assembly with an initial enrichment of 2%, a BU of 40 GWd/tU and a CT of 2 years.

ORIGEN-ARP offers modelling of several types of fuel. To be consistent with previous simulations, the 8x8 BWR fuel geometry was selected in this work. This geometry was also used in the second step, the simulations of the gamma-ray interaction, which were performed using the Monte Carlo software tool Geant4.

In the Geant4 simulations performed here, $10^5$ gamma photons were isotropically generated per fuel rod according to the source distribution obtained in the former step, exemplified in figure 1. According to the Monte Carlo technique, each photon was traced individually whereby its interactions with the materials of the model geometry were governed by statistical properties. In this context, one may note that the Cherenkov photons are emitted in all directions and in this work no angular discrimination was performed among them, in accordance with the previous simulations [2]. The validity of this description should be studied in greater detail but is outside the scope of this work. The details of the Cherenkov photon generation and detection in Geant4 are not the focus for this paper and the reader is referred to ref [3] for more details.

The three parameters which mainly affect the magnitude of the Cherenkov light emission are the fuel's initial enrichment, its total burnup and its cooling time. A higher burnup corresponds to a larger power outtake from the fuel and hence more fission products that may initiate Cherenkov emission. Conversely, a longer cooling time allows for a larger fraction of these fission products to decay and their successively lower abundance implies a lower Cherenkov light emission with time. The impact of the initial enrichment is not as clear as that of burnup and cooling time, but the differences in fission yields of U-235 and U-238,

and for high burnups also Pu-239, do affect the fission product contents and hence the emitted Cherenkov light intensity.

In order to relate the new modelling results to the older ones [2], an initial enrichment of 2% was selected for the dominant part of the simulations. As modern fuel may comprise uranium enriched to 5%, enrichments up to that level were also simulated in a study of the dependence of the Cherenkov light emission on this parameter.

The dominant part of the investigations were performed of the complete fuel assembly containing 64 rods with the purpose to extend the previously obtained Cherenkov intensity values to both shorter and longer cooling times, as well as to extend them to other initial enrichments. In addition to this, it is desired to investigate the uncertainties of the data points and in the future, also study how the Cherenkov emission depends on fuel geometry and irradiation cycle. In this study, burnups of 10, 20, 30, 40 and 50 GWd/tU and cooling times of 0.25, 0.5, 1, 2, 3, 5, 7, 10, 15, 20, 30, 40 and 50 years were chosen.

Finally, one may note that the power level and irradiation history will also affect the Cherenkov light intensity, since it will affect the gamma-ray emission. For this reason, a limited study of how changes in the power level and irradiation history affects the light intensity is also included. Just like the other fuel parameters, the irradiation histories were initially also selected to be as similar as possible to the previous simulations. Accordingly, the power level during every cycle in each simulation was 8 MW/tU for 10 GWd/tU, 16 MW/tU for 20 GWd/tU and 24 MW/tU for 30-50 GWd/tU. The irradiation cycles were chosen to be 330 days followed by 35 days of outage, for all cycles except for the last one. Given the desired burnup for each simulation, the selected power production and the maximum irradiation time of 347 days chosen in the previous simulations, the last cycle became 260 days for BU=10-30 GWd/tU, 347 days for 40 GWd/tU and 104 days for 50 GWd/tU.

## 4. Results

### 4.1 Dependence on burnup, cooling time and fuel cycle history

In figure 2, the simulated Cherenkov-light emission for a complete 8x8 BWR fuel assembly with an initial enrichment of 2% is shown, together with simulated data from earlier simulations. The data sets are normalized to their respective mean values, taking into accounts only data points that are common for both cases i.e only including cooling times of 1-40 years.

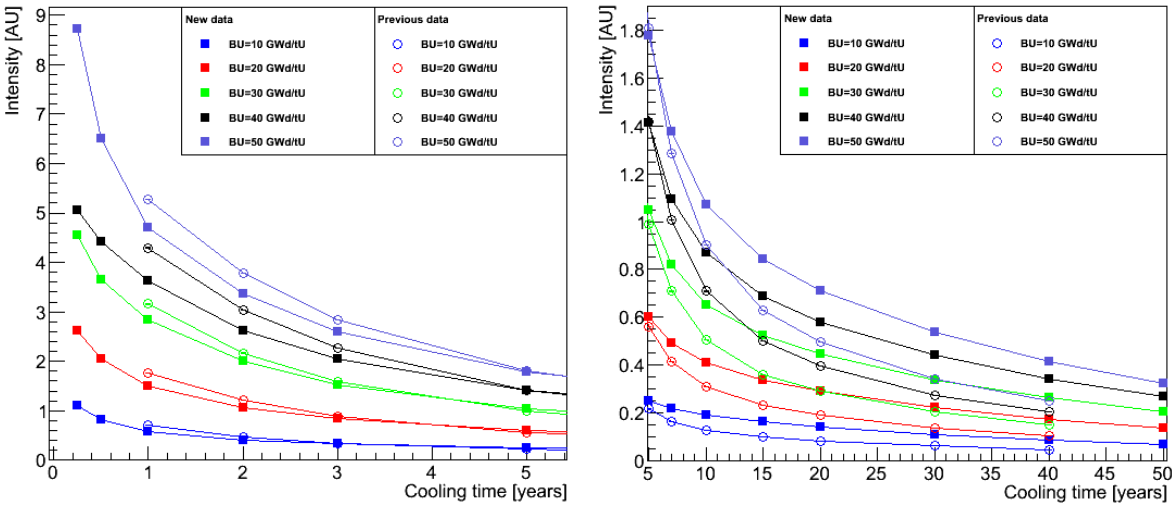


**Figure 2.** Simulated relative Cherenkov light emission from a 8x8 BWR fuel assembly with an initial enrichment of 2%. The squared markers indicate the new results, the circles mark original results.

No comparison between absolute emission values between the new and previous simulations has been performed for the simple reason that previous data do not provide absolute numbers. However, with the current normalization, it is seen that the simulated Cherenkov-light emission falls off slightly slower with cooling time as compared to the previous data. As compared to previous simulations, the new simulations have been extended to shorter cooling times, which show a much higher rate of Cherenkov-light emission. The intensity after 3 months cooling time, as compared to one year, is almost 70% higher for a high-burnup of 50 GWd/tU fuel and almost 60% for 10 GWd/tU fuel. However, the corresponding Cherenkov intensity for BU=40 GWd/tU shows a significantly smaller increase. One identified possible reason for this may be the irradiation history, because the last irradiation period simulated for this specific burnup is longer than in the other simulations.

In order to get an indication of how much the irradiation history may affect the Cherenkov emission, two additional simulations were performed. Now, the power production was chosen such that the last irradiation period amounted to 260 days for both BU=40 GWd/tU and BU=50 GWd/tU. The results at a CT of 3 months are shown in table 1 and reveal that the irradiation history does in fact matter. Adjusting the number of days in the last cycle gives a change in Cherenkov emission of 2-3%.

| BU [GWd/tU] | Power production [MW/tU] | Number of irradiation cycles | Days in the last irradiation cycle | Relative Cherenkov emission intensity |
|---|---|---|---|---|
| 40 | 24.0 | 5 | 347 | 5.06 |
| 40 | 21.0 | 6 | 260 | 5.17 (+2.2%) |
| 50 | 24.0 | 7 | 104 | 8.73 |
| 50 | 22.3 | 7 | 260 | 8.49 (-2.7%) |

**Table 1.** Relative Cherenkov emission intensities at a CT of 3 months in the “default” simulation cases for BU=40 GWd/tU and BU=50 GWd/tU as well as for a selected case when the last irradiation cycle has 260 days, in accordance with the simulations of lower burnups.

Although it is not visible in figure 2, the new data points do include statistical uncertainties as estimated by Geant4 based on the stochastic nature of Cherenkov light emission. These uncertainties are in the order of 0.07%, irrespective of burnup and cooling time. Repeating the same simulation four times for the data point with BU=30 GWd/tU and CT=20 years gives a maximum variation of around 0.2% and a standard deviation of 0.1%, confirming this level of precision. However, other sources of uncertainties and approximations are expected to cause significantly larger imprecision, such as the irradiation history for which the impact of needs to be individually modelled for each (authentic) fuel assembly.

Due to the lack of short cooled 8x8 fuel in the world, depending on that this fuel type is not manufactured any more, it is not possible to collect new data with cooling times below 10-15 years. And although scattered data exist from earlier measurements, it is not enough to distinguish which of the two sets of results are most accurate in their description. Instead, a dedicated experimental study is suggested, according to the following section.

## 4.2 Dependence on initial enrichment

The dependence of the emission of Cherenkov light on the initial enrichment was investigated at a BU of 20 GWd/tU and cooling times of 5 and 30 years. Simulations were performed for initial enrichments of 3%, 4% and 5%, in addition to the 2% already covered in the previous section. It may be noted that the neutron flux is implicitly specified in Origen-ARP in terms of the fuel burnup, which means that for the same burnup but different initial enrichments, the neutron flux is adapted so that the total amount of gamma-rays produced will differ.

The results are shown in figure 3. The lowest intensity is obtained with the highest initial enrichment of 5%. The effect of the enrichment on the Cherenkov emission is strongly dependent on time. The

Cherenkov emission for an enrichment of 5%, a BU of 20 GWd/tU and a CT of 5 years is for instance 13% lower than the emission for 2% enrichment. For a CT of 30 years, the corresponding difference is 0.16%.

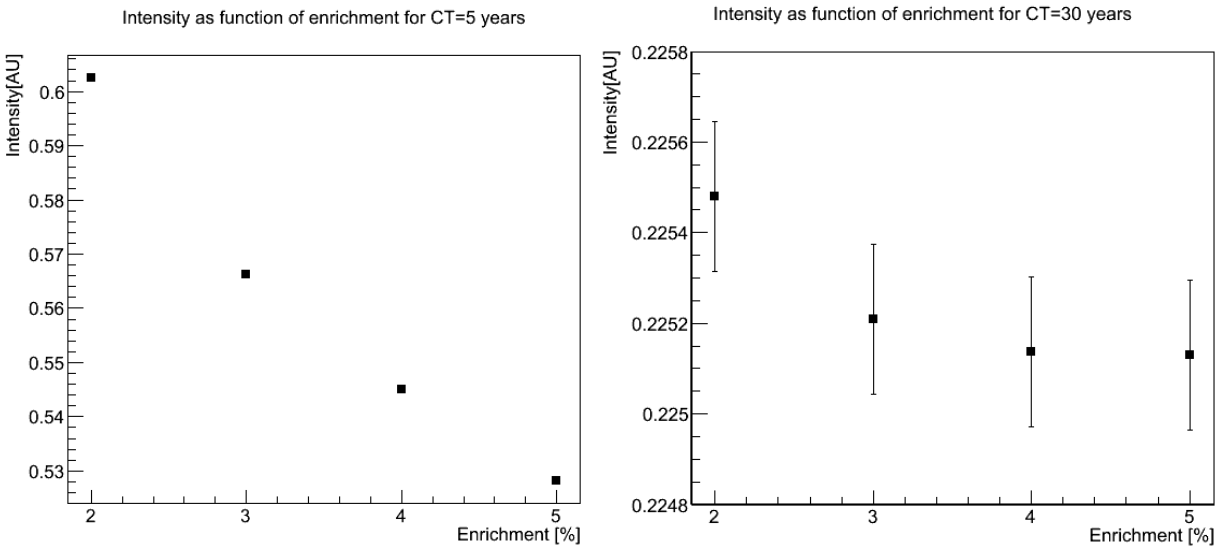


**Figure 3**. The Cherenkov light intensity for initial enrichments of 2-5% for a BU of 20 GWd/tU and a cooling time of 5 years (left) and 30 years (right). The apparent difference in error bars is due to the use of different scales on the vertical axes in the figures. Note that both axes have been cut.

One explanation to the decrease in Cherenkov emission intensity with increasing initial enrichment may be found in the gamma ray spectrum created in Origen-ARP. More detailed results from the simulations of a CT of 5 years are presented in table 2. There it is shown that the total number of events in the input spectrum to Geant4 decreases with increasing enrichment, most likely due to the adaptation of the neutron flux in Origen-Arp (in order to reach the specified burnup and power production) which in turn affects the interaction in the fuel and hence the gamma-ray spectrum.

| Initial enrichment [%] | Total number of events in Origen-Arp spectrum [$10^{18}$] | Cherenkov intensity [AU] |
|---|---|---|
| 2 | 2.30 | 0.60 |
| 3 | 2.22 | 0.57 |
| 4 | 2.17 | 0.55 |
| 5 | 2.13 | 0.53 |

**Table 2**. Simulated data for assemblies with a BU of 20 GWd/tU, a CT of 5 years and initial enrichments of 2%, 3%, 4% and 5%.

A deeper study of the connection between the gamma-ray source spectrum, the electron energy distribution and the resulting number of Cherenkov photons is needed in order to conclude the detailed features that give rise to these results. It should however be noted that initial enrichment does play a role in the Cherenkov emission.

## 5. Conclusions and outlook

The detection of Cherenkov light is one of the means used in the verification of spent nuclear fuel in accordance with nuclear safeguards agreements in force. In order to draw conclusions on partial defects where a fraction of the fuel material has been removed and/or replaced, using the Cherenkov light, one

must have a predictive capability accurate enough to enable experimental verification within decent limits. The higher the precision of predictions as well as measurements, the smaller diversions may be detected. At present, a sensitivity to detect 50% partial defects is established, but improved procedures may lead to a lowering of this limit.

Earlier simulations have been performed [2], but without being experimentally confirmed. It is however possible to repeat the simulations using other software in order to get an estimate of their accuracy. In this case, the repeated simulations result in differences in the relative Cherenkov light emission of -18% for BU=10 GWd/tU and one year cooling time, and +80% for the same burnup but at 40 years cooling time. The results point out a need of experimental data, for the purpose of systematically studying the response of the DCVD to irradiated nuclear fuel with varying fuel parameters and irradiation history.

From the results, it can be seen that cooling times shorter than one year result in a very high Cherenkov emission. It can also be seen that the dependence of the Cherenkov light emission on fuel burnup and cooling time from the older simulations is not fully repeated. The fall-off of Cherenkov emission in the new simulations is slightly less steep with increasing cooling times.

In addition, it has been shown that there is a dependence on Cherenkov light emission on the initial enrichment, in particular at short cooling times, explained by the fact that the enrichment affects the gamma-ray spectrum. The details of the reasons for this dependence may be focus for future studies.

A number of items may be subject for future work:

- In order to obtain experimental verification for the results, the next step is to simulate other fuel geometries, such as common PWR geometries, which are available for verification also for cooling times shorter than 10 years.
- In connection to this, it is also desirable to study if and to what extent a single fuel rod can be simulated instead of a full fuel assembly, in order to speed up the simulations.
- In addition, because experimental measurements have pointed out large variations in detected Cherenkov light intensity for spent nuclear fuels with almost identical burnup but different irradiation histories, investigations of this effect have a high priority.
- An experimental campaign should be performed in line with the above suggestions,
- The apparent reduction in Cherenkov light intensity for increasing initial enrichment should be studied.
- Furthermore, in this work, the total emission of Cherenkov light has been studied. Since safeguards verification using the DCVD is performed using instrumentation placed above the fuel, only Cherenkov light emitted in the vertical direction is collected. Further studies have to be made to investigate whether the vertical component of the light may be represented by the total emitted light in order to apply these data to the analysis of fuel inspection data.

The new Cherenkov emission intensities will be valuable in the continuous process of developing the capabilities of the DCVD as a partial defect tester. By improving the predictive capabilities using accurate modelling, the capabilities to detect smaller fractions of diverted fuel material may also be expected to improve. This work discusses a first and very important step on the way, where further verification of other fuel types and irradiation histories is crucial.

## 6. Acknowledgements

The Geant4 computations were performed using resources from Uppsala Multidisciplinary Centre for Advanced Computational Science (UPPMAX) under project p2007011.

Thanks also to the Swedish and Canadian IAEA support programmes for information sharing and fruitful discussions.

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# Geology for the country report

**Alexandra COLIN[1], Danièle BUIGUES[1], Jean-Laurent BERNARD[1], Michaël LALOUA[2]**
**1-Commissariat à l'Energie Atomique; Centre DAM Île-de-France,**
**Bruyères le Châtel, 91 297 Arpajon Cedex, France.**
***alexandra.colin@cea.fr.***
***daniele.buigues@cea.fr***
***jean-laurent.bernard@cea.fr***
**2- GeoPlusEnvironnement, 31290 Gardouch, France**
***m.laloua@orange.fr***

## Abstract

A country report is one pivotal element of the assessment of a specific country regarding nuclear non-proliferation. It aims to provide detailed information on nuclear related activities of the country under assessment, such as nuclear development, scientific and technical capabilities, fissile material and technical resources, etc. Country reports are mainly based on open sources. Ascertaining the origin, the nature and the credibility of information and data are essential in particular for open sources. Then, information has to be merged and expertise to be applied. Several performing tools are routinely used to understand the situation and evaluate the ability of fissile material production, like cartography, imagery, analysis of scientific publications (bibliometry), etc. However an interesting aspect of the nuclear capabilities of the country is its potential self sufficiency in raw material (uranium resource) which cannot be fully estimated with those classical tools only. In order to carry out this evaluation, we have considered a new tool based on geology through the following types of data:

- geological and metallogenic data (maps),
- IAEA geological types of uranium deposits and the NEA/IAEA classification scheme for uranium resources,
- bibliographic studies about uranium resources (prospecting, recovery, uranium grade, etc.),
- open sources from geological institutions (websites, etc) ,
- geologists' analysis,
- other field factors like access to the site, uranium extraction process, etc.

When all those data have been collected and assessed, criteria are extracted which could be combined in order to obtain the most important uranium target and so to infer knowledge about a potential area for uranium extraction in a specific country.

Geology appears to be a perfect "tool" to determine potential uranium resources with precise location and to evaluate the potential of uranium self-sufficiency of a country. Afterwards, it is possible to compare the results with information from other sources and complete them. An example of the application of this tool to one country is described in this paper.

## 1- The new tool in the context of a country report

A country report is mainly based on open sources. Ascertaining the origin, the nature and the credibility of data and information is essential to the reliability of the report. Even if some performing tools are routinely used to understand the situation and evaluate the ability of fissile material production (like cartography, imagery, analysis of scientific publications, etc.), this is sometimes not enough. The question we have to answer is: "*does this country present a risk of proliferation?*" We have to analyse the capabilities (both human and technical), the nuclear fuel cycle from mines to (a possible) weapons program and the mastering of missile technology as possible means of delivery.

Two cases have to be considered:

- The country has already a nuclear civil program. Then, we carefully assess the possibilities of fissile material diversion from civilian to military uses (through undeclared activities and facilities).
- The country wishes to develop a nuclear civil program. Then, attention should be paid to the progress of the project.

We need to evaluate the level of scientific and technical/industrial capabilities of the country (know-how, equipment, material, etc.). Attention is paid specifically to facilities as enrichment or reprocessing plant which can produce enriched uranium and/or plutonium. We also focus on the mineral resource potential. Indeed, it is worth knowing the country's uranium production capabilities: mining, milling, purification and conversion on which this analysis could provide indications on the degree of country natural uranium's self sufficiency. As uranium concentrates production potential is essential to feed the head of the nuclear fuel cycle, it constitutes important information. In that goal, we developed a specific tool to know easily it and its precise locations.

## 2- Method

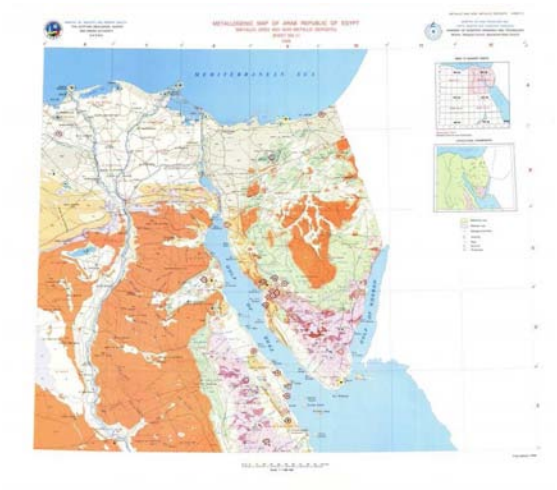
We have considered a new tool based only on geology through different types of data:

- Geological and metallogenic data

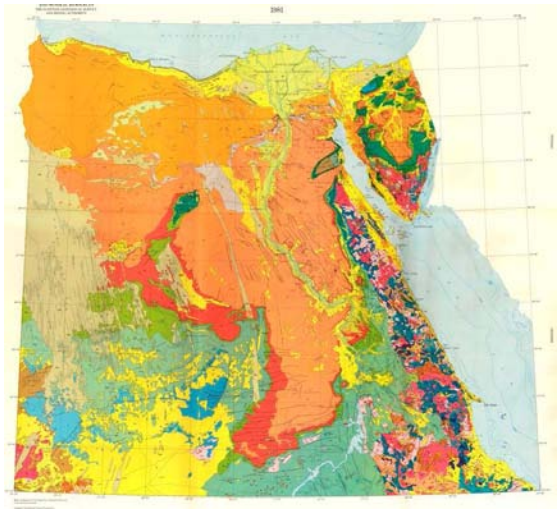
For each country, all the geological and metallogenic maps and data are collected (cf. figures 1 & 2, example of geological maps for Egypt).The knowledge of the formation of the geological deposits and their genesis is essential. The consequence is the presence or absence of uranium. We have to know the exact mineralogy because uranium could be found in two types of configuration: as a product or as a sub-product (of phosphates, gold, rare earth, etc.). This information is the first to be known because it will give indication on the process of extraction

(cost, feasibility, etc.).

*Example of geological maps for Egypt:*



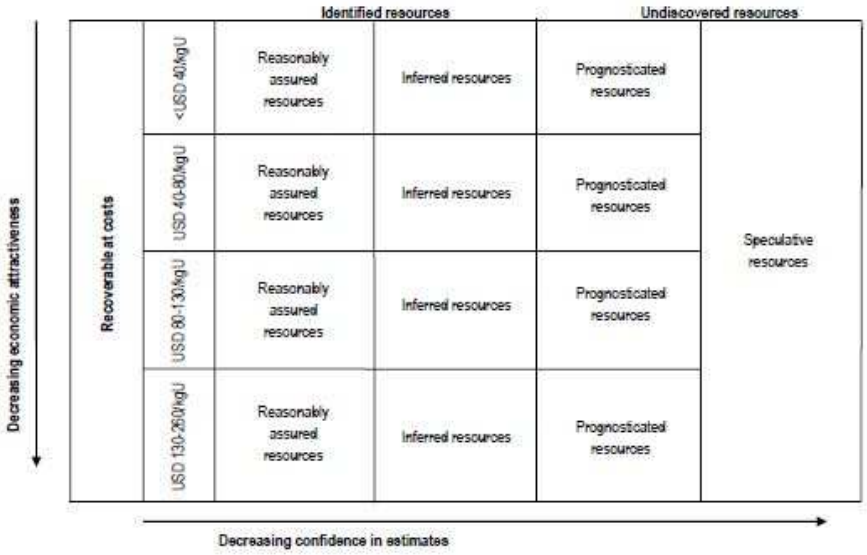
*1-Metallogenical map of Egypt*
*Credit: ministry of industry and mineral wealth, the Egyptian Geological Survey and Mining Authority and ministry of high education and state ministry for scientific research, Academy of scientific research and technology, 1998*



*2- Geological map of Egypt*
*Credit: Ministry of industry and mineral resources, The Egyptian Geological Survey and Mining Authority, 1981*

- IAEA geological types of uranium deposits and the NEA/IAEA classification scheme for uranium resources

The uranium deposits and resources are classified by the IAEA. It will be important to establish if the resources are reasonably assured, prognosticated or speculative (cf. figure 3). We will try to find the type of prospection (radiometric, excavation, drilling, etc.). A lot of countries declare uranium resources to the IAEA [1]. These declarations will be compared to the results at the end of our study.




*3-NEA/IAEA Classification scheme for uranium resources*
*Credit: IAEA, 2007*

- <u>Bibliographic studies about uranium resources</u>

We try to find all the bibliographic studies on geology of the considered country (cf. fig 4). The studies could have carried out by the country itself or by other countries.

We evaluate the uranium quantities, locations, grades and the method for the prospection. All references by years and by authors are collected.

It is possible to know when the country has begun its search, how, who, etc. to evaluate the level of knowledge and the progress.

**Lithologic Discrimination Using Selective Image Processing Technique of Landsat 7 Data, Um Bogma Environs Westcentral Sinai, Egypt**

**Adel Zein Bishta**
*Faculty of Earth Sciences, King Abdulaziz University, Jeddah, Kingdom of Saudi Arabia*
*E-mail: azbishta@hotmail.com*

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Abstract. Um-Bogma area located in the westcentral part of Sinai, Egypt is covered by Precambrian basement rocks within the northern part of the Precambrian Arabian-Nubian crystalline massive. Satellite images are among continuous sources of digital data for mapping lineaments and lithology. In the present work, the lithologic discrimination and lineament analysis of Landsat-7 ETM+ data for Um Bogma area have been carried out using Geomatica PCI EASI/PACE (9.1) and GeoAnalyst PCI packages.

A selective image processing technique (SIPT) is a new approach in Geomatica, which gives rise to valuable results in this work. The SIPT and the visual interpretation of the constructed Landsat false colour composite (FCC) and ratio images led to discriminate and delineate the lithologic rock units of the study area.

The structural lineament pattern for each interpreted lithologic unit has been constructed using SIPT. The prepared rose diagrams for the extracted lineaments are very characteristic for each interpreted lithologic unit. The structural lineament analyses and the discriminated lithologic units are integrated to show their relationships to the high radioactivity occurrences of the study area. The NW-SE trend is the predominant regional structural lineament trend in the investigated area. The NE-SW to the ENE-WSW directions is the predominant minor structural lineament trends. The highest radioactivity zones of the study area are restricted to Um Bogma formation and the younger granite rocks.

*3- Example of scientific publication*
*Credit: Faculty of earth sciences, king Abdulaziz University, Jeddah, kingdom of Saudi Arabia*

- Open sources and expert analysis

We also establish an inventory of the active uranium mining location, the actors in the country for the mining activity, the entities who could exploit it (sometimes other countries); etc. We also look for information (data and pictures) from local geological institutions. Project's presentation could have been made in congress or open meetings with the IAEA or with other countries. We have also setup a geologist expert network who could provide information and help us to complete our data if it happens they went to a country of interest.

Access to the sites and their possible restriction (touristic or not for example) are also considered.

*5-Picture of a detailed view of mineralized structure during prospection*
*Credit: photograph's expert*

For some countries, we have no information from international geologists or national geologists in scientific publications. Nevertheless, we could evaluate areas where uranium may be located without any data (quantities, grades, etc.). But it is not a frequent case.

## 3- Results : the tool

When all the data have been gathered and assessed, they are classified on the base of **10 possible criteria**. From this data we have enough information to answer specific question about resources and/or location.

The pertinent criteria are selected according to the objectives of the study (cf. table 1)

| Area/Site | Uranium bearing mineral | Prospection | Grade | Quantities | Ore extraction | Uranium recovery | Uranium alone or affiliated | Access to the site | Access Restriction |
|---|---|---|---|---|---|---|---|---|---|
| Name | Monazite, etc. | Radiometric research, etc. | ppm U | Tons | Feasibility depends on the mineralization geometry (vein, etc.) | Easy process, etc. | With gold, coal, etc. | Road, etc. | Touristic site, etc. |

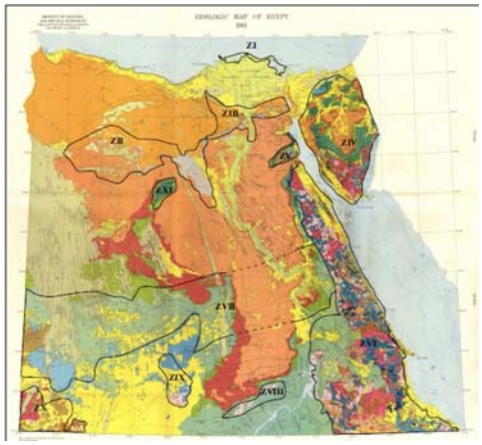
*Table 1: criteria*

Different types of information could be drawn from the application of the criteria:

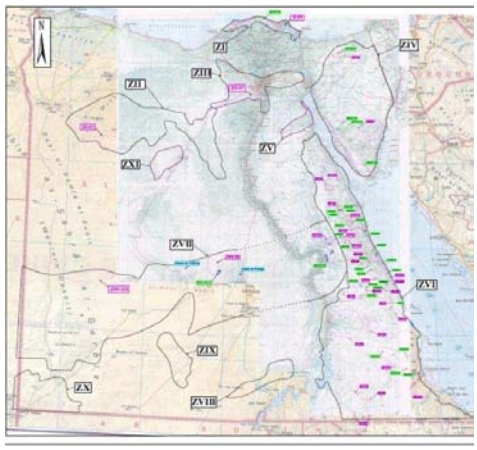
- the total quantity of the uranium resources,
- the general areas where uranium could be found. Results are expressed on a map,
- the precise location site by site is also possible. Specific information on the site (grades, quantities, access, etc.) is gathered in an Identity card (ID).

## 4- Example of Egypt

The exercise was made on the case of Egypt. We obtained 11 areas of interest and 54 precise sites where uranium could be found, so we made 54 ID cards for them.




*6- Map with 11 areas*
*Credit: CEA*



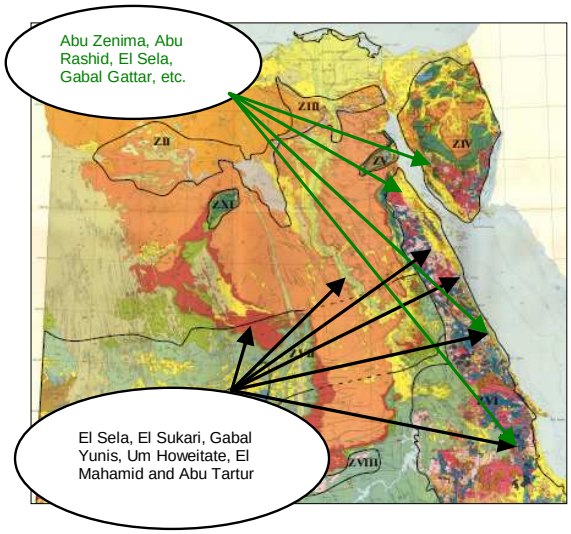

*7- Map with 54 precise sites*
*Credit: CEA*

| District ZVI-D1 : Abu Rushied | | |
|---|---|---|
| District | | ZVI-D1 = Abu Rushied |
| Localisation | | 60 km au SSW de la ville de Marsa Alam<br>De 24°45' de latitude Nord à 24°35' de latitude Sud<br>De 34°40' à 34°50' de longitude Est |
| Accès | | Depuis la mer Rouge via les vallées de Wadi Gamal, Wadi Nugrus puis Wadi Abu Rushied |
| Superficie | | 4 à 8 km² |
| Prospection | | Cartographie géologique avec une précision de 20 m par la NMA<br>11 tranchées par la NMA<br>Sondages par une compagnie canadienne |
| Type de minéralisation | | Type 5 (Minéralisation filonienne) et type 8 (métasomatique) |
| Paragenèse minérale | Minéraux accessoires d'origine du granite source d'U | Actinolyanite<br>$(Ca_2(REE)_2(UO_2)(CO_3)_5 (OH)_2.5H_2O)$<br>Betafite<br>$((Ca,Na,U)_2 (Nb,Ta)_2O_6.nH_2O)$ |
| | Minéraux uranifères primaires | Pechblende ($U_3O_8$) |
| | Minéraux uranifères secondaires | Uranophane : ($CaO.2UO_3.2SiO_2.6H_2O$)<br>Kasolite : $[Pb(UO_2)(SiO_4)(OH)_2]$<br>Torbernite : $[Cu (UO_2)2(PO_4)_2.8-12H_2O]$<br>Autunite : $[Ca(UO_2)_2 (PO_4)_2.3H_2O]$ |
| | Autres minéraux | Pyrite, oxydes de fer, minéraux zincifères et manganésifères |
| Modèle métallogénique | Source | Brèches et mylonites issues du granite potassique calco-alcalin à alcalin d'Abu Rushied |
| | Vecteur | Fluide hydrothermal acide et alcalin |
| | Piège | Chimique : l'intrusion de lamprophyres peralumineux et calco-alcalins à caractère réducteur conduit à la précipitation des minéraux uranifères primaires (pechblende) |
| Teneurs | Source (Brèches et mylonites issues du granite potassique calco-alcalin à alcalin d'Abu Rushied) | 1 à 10 ppm U / 3 à 15 ppm Th / jusqu'à 1000 ppm Nb et 300 ppm Ta dans le granite |
| | Lamprophyres | 440 ppm U / 40 ppm Th en moyenne |
| | Conduit Lamprophyres/ calcaires | 1500 ppm U / 1700 ppm Th en moyenne |
| Typologie de ressources estimées | | Ressources prognostiquées |
| Estimation des ressources | Ressources estimées et publiées | Néant |
| | Ressources estimées par G+E | 800 t d'U / 160 t de Th |
| Sous-produit / coproduit | | De l'exploitation du niobium et du tantale dans le massif granitique d'Abu Rushied |
| Extraction du minerai | | Coûteuse : volumes importants à mobiliser pour peu de minerais (petites veines en lamprophyres dans un encaissant granitique) |
| Recuperation de l'uranium | | L'uranium est contenu dans la pechblende ou dans des minéraux uranifères hexavalents, le process d'extraction de l'uranium est connu et facilement maîtrisable par les égyptiens |
| Typologie du traitement (G+E) | | Lixiviation acide en tas : type C1 |
| Objectif retenu | | District ZVI-D1 |
| Minéralisation | | Pechblende |

8- *ID card for one site*
*Credit: CEA*

We can compare our results with information from newspapers for example (cf figure 9):

- in 2011, Egypt announced 6 potential industrial sites without any precision on the location. By applying 4 selected criteria that we consider as industrially strategic (easy access, no restriction, high uranium quantity, easy ore extraction), **then 6 sites of interest have been determined**. However, we were not able to confirm if our results corresponded to the Egyptian sites. Indeed, we could have chosen other criteria (as grades, etc.) and then find more or less other sites.
- in January 2013, Egypt mentioned **4 names** with 2 important criteria (high concentration and quantities). Through our application, with these two criteria, we found 16 sites including the 4 sites announced officially.




*9- Location of sites in 2011 and 2013*

*Credit : CEA*

In the newspapers, some criteria were missing or not precise, but nevertheless we were satisfied with our results.

## 5- Conclusion

Geology has been proven to be a very useful “tool” to carry out the assessment of uranium resources for specific countries. It provides precise areas and sites locations of these resources. Moreover, it is versatile and allows an easy comparison and updating of information. For some countries, the tool is used differently, with less precision because of the lack of data, but it can give some interesting information as the credibility of presence of uranium.

## 6- References

[1] Nuclear Energy Agency; Uranium 2011: resources, production and demand; OECD; 2012

# Development of a reference spent fuel library of 17x17 PWR fuel assemblies

**Riccardo Rossa[1,2], Alessandro Borella[1], Klaas van der Meer[1]**

[1] Studiecentrum voor kernenergie–Centre d'etude de l'energie nucleaire (SCK•CEN)
[2] Université libre de Bruxelles (ULB)

Email: rrossa@sckcen.be

***Abstract:***

*One of the most common ways to investigate new Non-Destructive Assays (NDA) for the spent fuel assemblies are Monte Carlo simulations. In order to build realistic models the user must define in an accurate way the material compositions and the source terms in the system.*
*This information can be obtained using burnup codes such as ORIGEN-ARP and ALEPH2, developed at SCK•CEN. These software applications allow the user to select the irradiation history of the fuel assembly and to calculate the corresponding isotopic composition and neutron/gamma emissions as a function of time.*
*In the framework of the development of an innovative NDA for spent fuel verifications, SCK•CEN built an extensive fuel library for 17x17 PWR assemblies, using both ORIGEN-ARP and ALEPH2. The parameters considered in the calculations were initial enrichment, discharge burnup, and cooling time. The combination of these variables allows to obtain more than 1500 test cases.*
*Considering the broad range of the parameters, the fuel library can be used for other purposes apart from spent fuel verifications, for instance for the direct disposal in geological repositories.*
*In addition to the isotopic composition of the spent fuel, the neutron and photon emissions were also calculated and compared between the two codes. The comparison of the isotopic composition showed a good agreement between the codes for most of the relevant isotopes in the spent fuel. However, specific isotopes as well as neutron and gamma spectra still need to be investigated in detail.*

**Keywords:** Spent Fuel; PWR; ORIGEN-ARP; ALEPH2

## 1. Introduction

Spent fuel is characterized by a very high neutron and gamma emission due to the radioactive decay of the isotopes that have been produced during irradiation. The irradiation in the reactor produces numerous fission products and actinides that make the resulting isotopic composition and associated gamma spectra particularly complex [1]. Considering the high neutron and gamma field together with the significant heat coming from the decay of the fission products and actinides, the spent fuel is generally stored under water in a spent fuel pool and it is not directly accessible by the safeguards inspectors. Given the characteristics listed above, the spent fuel can be considered as one of the most difficult materials to verify during an inspection.
Non-destructive assays (NDA) are one of the possible methods to verify the spent fuel and several research projects are trying to improve their current capabilities [2],[3]. One of the ways to investigate new NDA for the spent fuel assemblies are Monte Carlo simulations. In order to build realistic models the user must define in an accurate way the material compositions and the source terms in the system. Therefore an effort was made to define a spent fuel library that serves as reference to benchmark the performances of the measurements methods under investigations.
The goals of this preliminary work are therefore twofold. The first one is to understand how the irradiation history of the fuel influences its isotopic composition (and consequently its neutron and gamma source strength). The second one is to generate automatically several input cards compatible with the MCNPX [4],[5] code that will be used in the study of innovative NDA techniques. To achieve both goals the spent fuel needs to be characterized in terms of isotopic composition, neutron and gamma emission (both source intensity and energy spectrum).

## 2. Computational models used for the calculations

The study of the time evolution of the nuclear fuel generally requires the combination of computer codes to model the neutron transport in the reactor and to predict the fuel composition due to the radioactive decay of its isotopes.
Two different codes have been used to perform the simulations: ORIGEN-ARP and ALEPH2.2.
The first code uses a set of pre-compiled averaged cross section values (called ORIGEN-ARP cross section library) as input to the depletion calculation. This procedure avoids the time consuming neutron transport and therefore leads to the strong decrease in the computational time required for the simulations.
The second code uses a statistical approach (Monte Carlo method) for the calculation of the neutron transport and then performs the fuel depletion. By using the Monte Carlo code MCNP(X), ALEPH2.2 retains the great flexibility in the definition of the system (geometry, materials) and in the nuclear data used in the simulation. As a drawback the computational time can be significantly higher than in the case of ORIGEN-ARP.

### 2.1. ORIGEN-ARP

The first code used to generate the fuel library is ORIGEN-ARP [6] and it is part of the SCALE package [7] that is developed by Oak Ridge National Laboratory (ORNL).
ORIGEN-ARP employs a graphical user interface (GUI) to select several characteristics of the fuel assembly. The PWR 17x17 (w17x17) case has been chosen considering its worldwide use in the nuclear power plants. The GUI is composed by several menus that are shown subsequently when all parameters in one window have been introduced and validated by the user.
Apart from the type of fuel geometry, the first window (*"Express"*) contains other parameters as follow:

- all results are normalized to 1 ton of uranium;
- the average power is set to 40 MW/tU. This value is taken from [8] and results in a burnup of 14.4 GWd/tU for a cycle of 360 days;
- the moderator density is the default value of 0.723 g/cm$^3$.

The next section is the *"Composition"* window:

- the abundances of the uranium isotopes are the default values proposed by ORIGEN-ARP depending on the initial enrichment selected in the previous section;
- oxygen was added to the list of isotopes in the fresh fuel in order to model the $UO_2$ material in the fuel pin. The natural isotopic composition was selected with a concentration of 134500 g/t.

The section *"Neutron"* allows selecting the energy-group structure for the neutron source spectra. The "238-group ENDF5" is the option chosen in the calculation. This group structure extends from 1E-5 eV up to 20 MeV. This section determines only the division of the energy groups and does not select the data library used by the code for the nuclear data.
At the same way, the photon emissions are treated according to a 74-group library defined by the user. The group structure extends from 10 keV up to 10 MeV.

The most extensive section in the ORIGEN-ARP interface is the *"Cases"* windows, where the user selects the characteristics of the irradiation history. All simulations for the fuel library consider irradiation cycles of maximum 360 days, after which the fuel assembly undergoes a decay period of 30 days. The duration of the last irradiation cycle is adapted to reach the desired level of burnup at the discharge. Following this procedure, the fuel assembly is exposed to the same power level during all cycles in the reactor. After the final unloading from the reactor (i.e. the last irradiation cycle is completed), the ORIGEN-ARP calculation considers 30 cooling times to compute the isotopic composition and both neutron and gamma emissions.

Other options selected in ORIGEN-ARP are:

- no cutoff is selected for the composition so all the isotopes are reported in the output file;
- output precision is 6 digits for the values of the mass concentrations;
- output precision is 4 digits for the neutron and gamma emission;
- the fuel matrix for the (alpha,n) evaluation is the $UO_2$;
- bremsstrahlung is not considered in the model.

### 2.2. ALEPH2.2

ALEPH is the Monte-Carlo burn-up code being developed by SCK•CEN since 2004 [9]. The code belongs to the category of shells coupling Monte Carlo particle transport codes and deterministic depletion algorithms.
The fuel assembly model considered in ALEPH2.2 tried to be identical to the case considered with ORIGEN-ARP. The geometry is the same of the PWR 17x17 fuel assembly modelled in ORIGEN-ARP (Table D1.A.3 of [6]), with the exception that the active length is reduced to 1 meter. It has been supposed that there is no water gap between neighbouring assemblies and this has been simulated placing reflecting surfaces around the fuel assembly.
The irradiation history is defined in terms of irradiation power (MW), length of the irradiation step (days), and length of the decay step (days, years). All these parameters are chosen to have the same conditions as those used for the ORIGEN-ARP simulations.
Other parameters in the input files are:

- the geometry of the fuel assembly (pin radius and pitch) and the initial composition of the fuel is the same of the ORIGEN-ARP simulations;
- the fuel temperature is 900 K;
- the cladding temperature is 620 K;
- the moderator temperature is 575 K (with a density of 0.723 g/cm$^3$);
- the water contains 630 ppm of Boron (according to abundances of $^{10}$B and $^{11}$B);
- each irradiation step runs a kcode with 5000 particles; initial value for $k_{eff}$=1.16; 100 cycles neglecting the first one from the calculations;
- the neutron source is modeled as a Watt fission spectrum with the parameters of the neutron-induced fission of $^{235}$U [4].

## 3. Main characteristics of the fuel library

The reference spent fuel library has been developed using three variables to take into account the most representative irradiation histories of the spent fuel. In particular:

- Initial enrichment (IE): 4 values from 3.5% to 5.0%, with increments of 0.5%
- Discharge burnup (BU): 14 values from 5 up to 70 GWd/tU, with increments of 5 GWd/tU
- Cooling time (CT): 30 values ranging from immediate discharge up to 3 million years

The combination of these parameters allows to obtain 1680 different compositions and source terms.

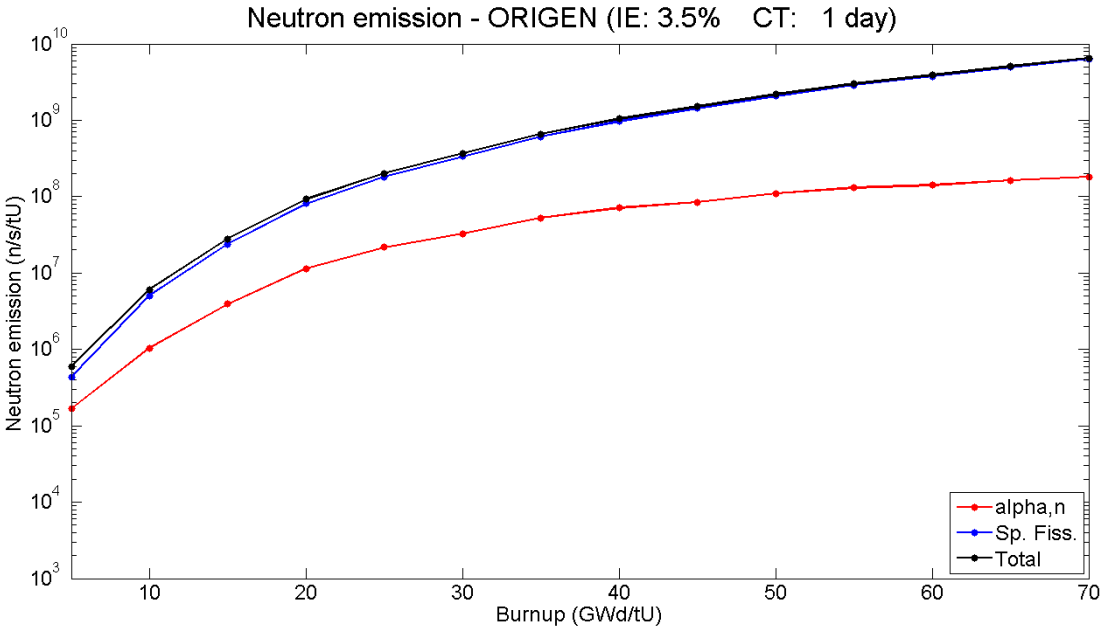



**Figure 1:** neutron emission as a function of the burnup.

Figure 1 shows the neutron emissions as a function of the burnup. The data have been calculated with ORIGEN-ARP, with a fuel of 3.5% initial enrichment and after a cooling time of 1 day. The plot reveals

that the spontaneous fissions are the main responsible for the neutron emissions after a short cooling time. This fact needs to be evaluated also with higher cooling times.
In order to do so, Figure 2 shows the neutron emissions as a function of the cooling time for the burnup value of 10 GWd/tU.
The contribution from alpha-n reactions is generally one order of magnitude lower than the term due to spontaneous fissions. The only exception is for burnup of 10 GWd/tU with cooling times higher than 30 years (Figure 2). For these combinations the two contributions are similar, with cases where the alpha-n reactions have the dominant role on the neutron emission.
Looking at Figure 2, there is a clear increase of the alpha-n contribution around 100 years of cooling time. This is due to the build-up of $^{241}$Am from $^{241}$Pu (half-life of 14.4 years). The same effect is present also to higher burnup, but it is not visible due to the higher neutron emission that conceals this contribution.

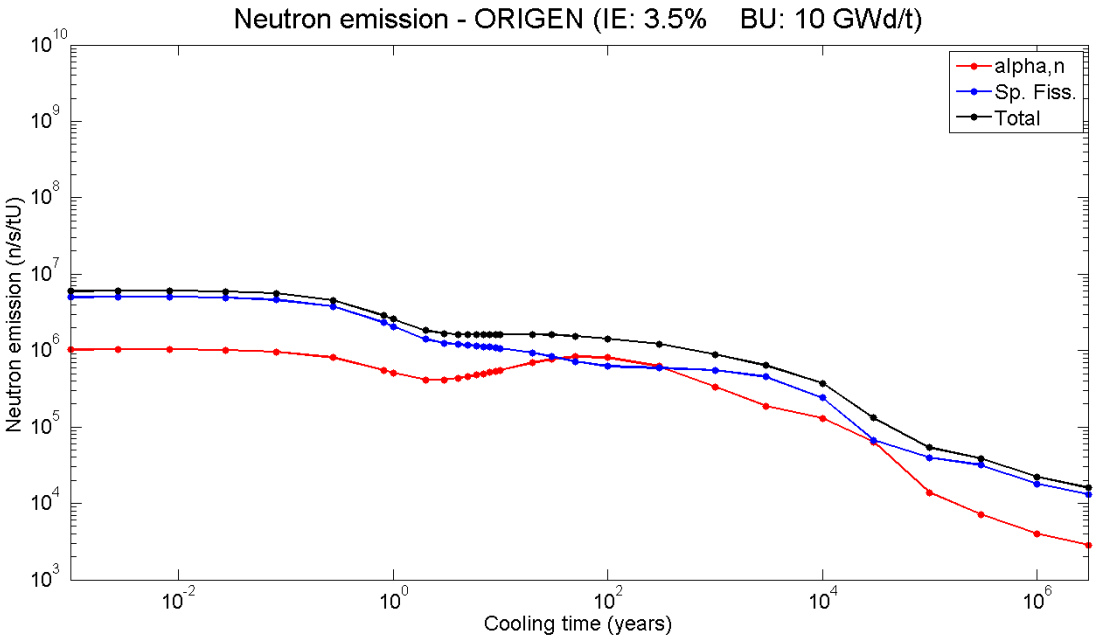



**Figure 2:** neutron emission as a function of the cooling time (BU: 10 GWd/tU).

In order to refine the study it is important to understand the role of each isotope in the total neutron emission. Figures 3 and 4 show the percentage of the total neutron emission due to each isotope as a function of the cooling time.

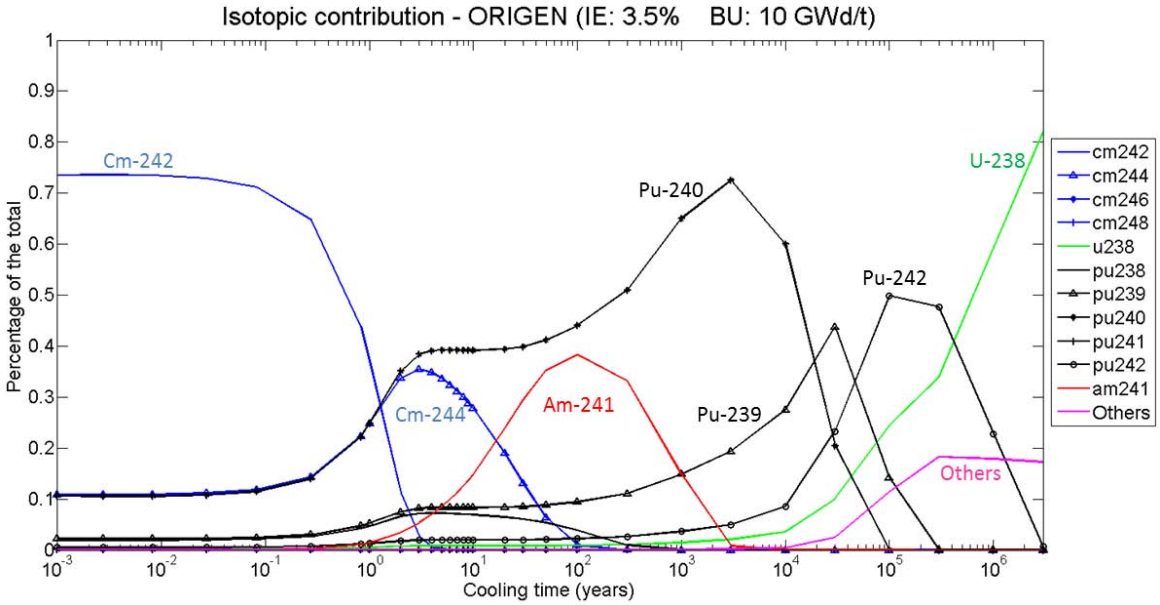



**Figure 3:** role of selected isotopes in the neutron emission (BU: 10 GWd/tU).

Both for Figure 3 and 4 there are 11 isotopes that are responsible for about 99% of the total neutron emission (the value 'Others' is relevant only for cooling times higher than 10000 years).
At low burnup the $^{242}$Cm is the main responsible for the neutron emission up to 100 days, role that is taken then by $^{244}$Cm and several plutonium isotopes. Another relevant contribution comes from $^{241}$Am (as it has been suggested from the peak in Figure 2). Other curium isotopes are not present at low burnup values because the short irradiation time did not allow the build-up of these high-Z isotopes.

With a burnup of 35 GWd/tU (Figure 4) two isotopes of curium ($^{242}$Cm and $^{244}$Cm) are responsible for basically all neutron emissions up to few years of cooling time but opposite to Figure 3 now $^{244}$Cm is the main actor. Increasing the cooling time another Cm isotope ($^{246}$Cm) has an impact on the total neutron emissions because the other previous isotopes have short half-lives. Other contributions come from $^{241}$Am, $^{240}$Pu, and $^{242}$Pu.
The latter isotope is the main responsible for the neutron emission in the long term, together with $^{238}$U, and a list of other isotopes (indicated as 'Others'). One must bear in mind that the magnitude of the emission for high cooling times is very low compared to the value at the discharge and this is the reason why many isotopes have a non-negligible role for long cooling times.

Increasing the burnup to 60 GWd/tU there are small differences to the case of 35 GWd/tU. $^{244}$Cm takes an even higher contribution for the short cooling times, as well as $^{246}$Cm for intermediate values. At very high cooling time $^{248}$Cm appears to be important along with $^{242}$Pu and $^{238}$U.

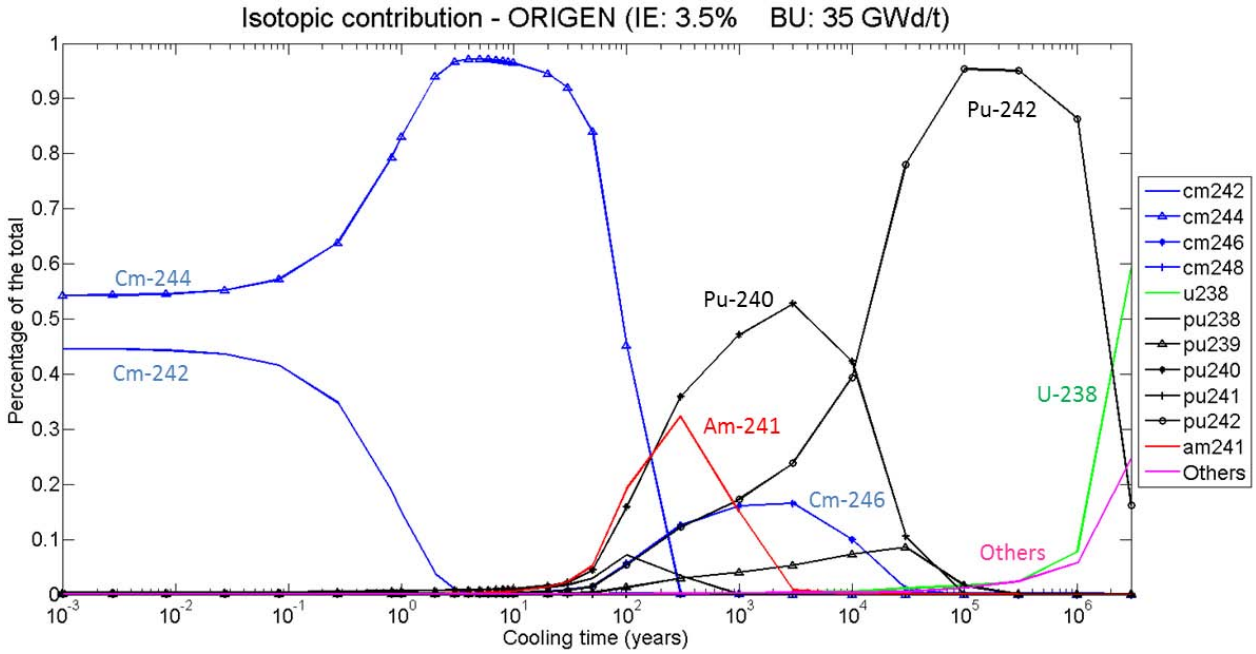



**Figure 4:** role of selected isotopes in the neutron emission (BU: 35 GWd/tU).

The analysis so far focused on the influence of burnup (BU) and cooling time (CT) on the neutron emissions. Since the initial enrichment (IE) plays also a role, the next part investigates this variable. Figure 5 shows the ratios of the total neutron emission as a function of the burnup for the enrichments considered in the simulations.
By looking at the magnitude of the difference between the enrichment values, it seems that this variable plays a minor role in the determination of the neutron emission compared to burnup and cooling time. In fact, while by changing the burnup the neutron flux value varies of more than one order of magnitude, the maximum difference with the initial enrichment is of less than a factor 3.
The origin of the peak observed in Figure 5 can be explained by looking at Figure 6. This plot shows the ratio of the total neutron emissions after a cooling time of 10 years. In addition to the curve 'Total' (that is the same of the curve '3.5%/5.0%' of Figure 5) also the ratios for specific isotopes are added. The curve related to the single isotopes is calculated taking the ratio between the neutron emission due to this isotope at 3.5% initial enrichment and the total neutron emission at 5.0%
It is clear that the shape of the curve 'Total' is determined by the one of $^{244}$Cm at burnup higher than 30 GWd/tU, while at lower burnup $^{239}$Pu, $^{240}$Pu, and $^{241}$Am determine the shape of the curve.

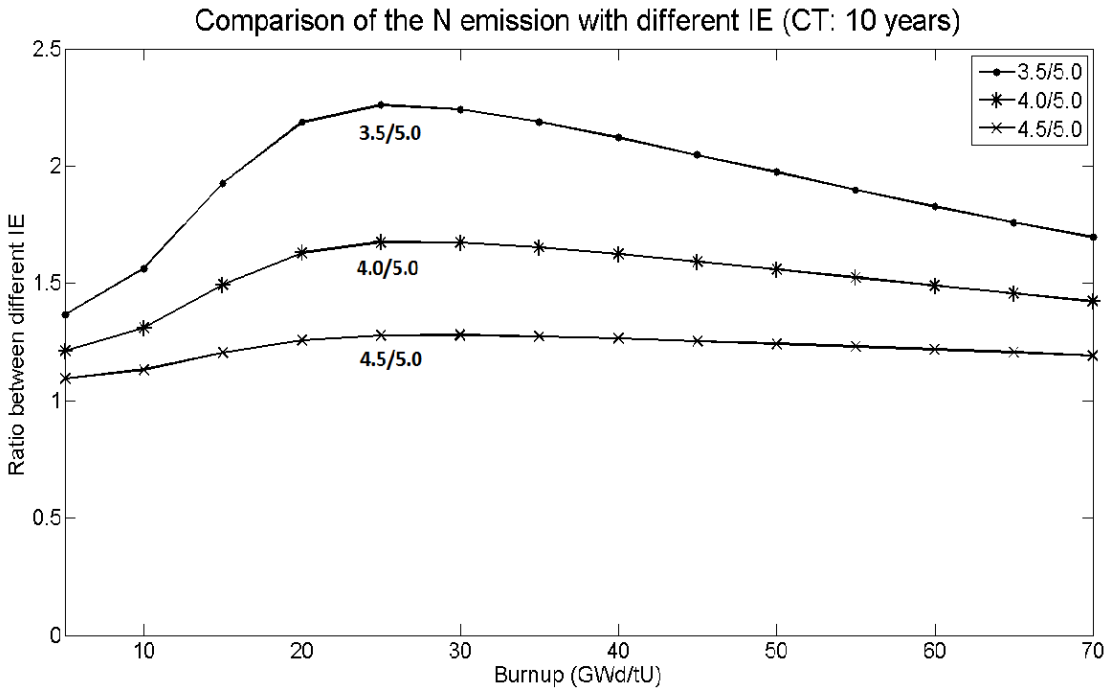



**Figure 5:** ratio of total neutron emissions for different enrichments as a function of the burnup (CT: 10 years).

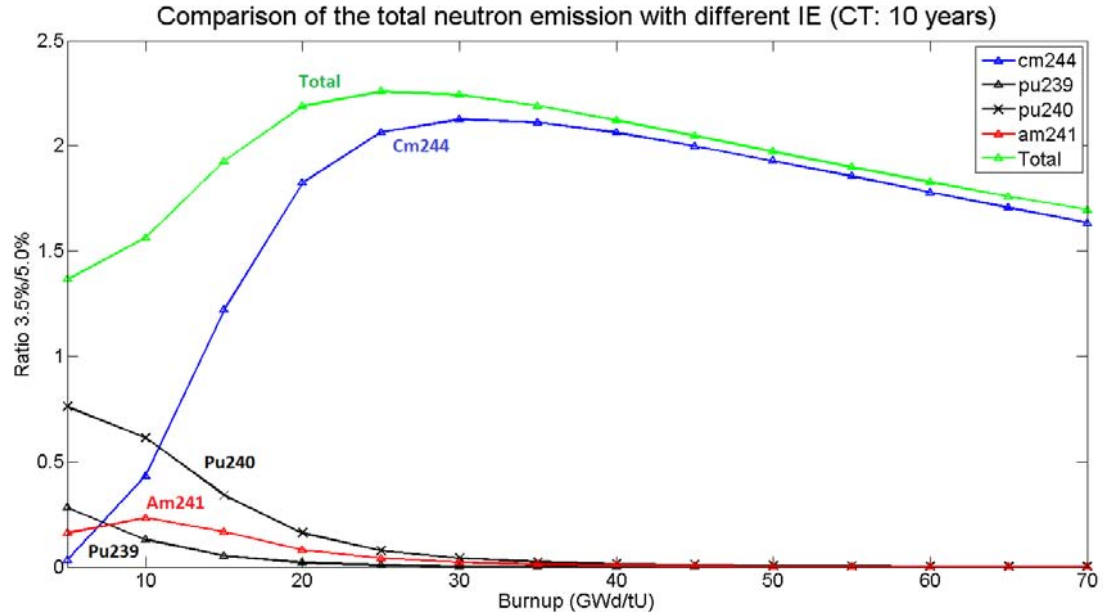



**Figure 6:** total neutron emission with different initial enrichments – role of selected isotopes (CT: 10 years).

## 4. Comparison of the two codes

The next step is to compare the two codes used to generate the spent fuel library. The set of simulations in the codes used the same values of:

- IE, BU, and CT;
- average power during irradiation;
- radius for the fuel pin, cladding, and pitch between neighbouring pins;
- boron concentration in the water.

The impact of the boron concentration, the water gap between the assemblies, and the data library on the final results will be investigated in the next section.

Each code applies a different normalization unit for the calculated data ($t_U$ for ORIGEN-ARP and $cm^3$ for ALEPH2.2). In order to compare the two codes it is necessary to renormalize one of the two to have the same measurement unit in all simulations.

Figure 7 compares the results of ORIGEN-ARP and the ones coming from ALEPH2.2 in relative terms. The fuel used in these simulations had an initial enrichment of 3.5%. ORIGEN-ARP always underestimates the emissions compared to ALEPH2.2. The ratio of the values coming from the two codes remains rather constant around 0.95 with the burnup. The only exception is at 100 years of cooling time, where it seems to be a decreasing trend for burnup higher than 40 GWd/tU.

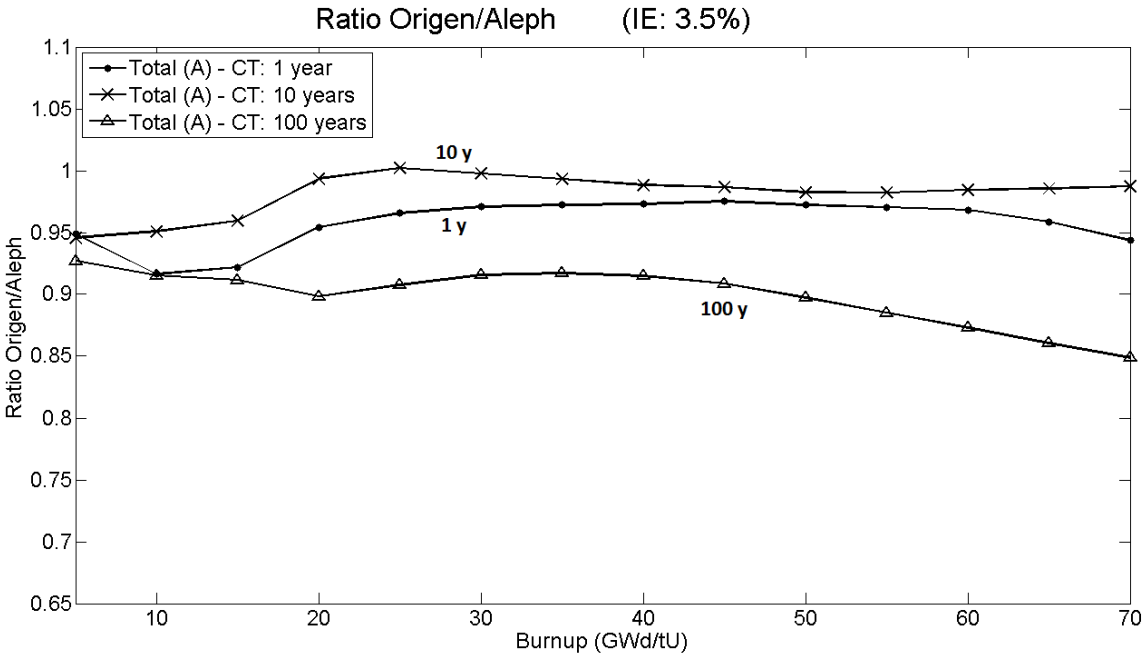



**Figure 7:** ratio of the neutron emission as a function of the burnup.

A comparison of the nuclides concentrations obtained by the two codes has been made in order to understand the reason of the difference in the neutron emissions. The following table shows the ratio between the concentrations of selected nuclides (the same list of Figure 3) as a function of the burnup. The isotopic compositions are calculated at direct discharge.
From Table 1 it is evident that ORIGEN-ARP underestimates the nuclides concentrations of the main neutron emitters compared to ALEPH2.2. This is then the reason of the lower neutron emission calculated with ORIGEN-ARP. The values in the table follow the same trend independently from the initial enrichment of the fuel. $^{242}Cm$ and $^{244}Cm$ (responsible for over 90% of the neutron emission up to few years of cooling time) are the isotopes with one of the lowest ratio between the two codes.

| Isotope / BU (GWd/tU) | cm242 | cm244 | cm246 | cm248 | u235 | u238 | pu238 | pu239 | pu240 | pu241 | pu242 | am241 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **5** | 0.862 | 1.126 | 0.875 | 0.870 | 0.996 | 1.000 | 1.101 | 1.012 | 1.024 | 1.046 | 1.091 | 1.052 |
| **10** | 0.828 | 1.005 | 0.740 | 0.697 | 0.991 | 1.000 | 1.046 | 1.000 | 0.997 | 1.063 | 1.071 | 1.049 |
| **15** | 0.840 | 0.978 | 0.698 | 0.645 | 0.987 | 1.001 | 1.023 | 0.995 | 0.976 | 1.068 | 1.079 | 1.079 |
| **20** | 0.857 | 1.007 | 0.732 | 0.669 | 0.981 | 1.001 | 1.004 | 1.000 | 0.990 | 0.993 | 1.054 | 1.076 |
| **25** | 0.845 | 1.007 | 0.737 | 0.693 | 0.976 | 1.001 | 0.985 | 1.001 | 0.991 | 0.990 | 1.018 | 1.057 |
| **30** | 0.838 | 0.998 | 0.724 | 0.696 | 0.970 | 1.002 | 0.966 | 1.004 | 0.988 | 0.991 | 1.003 | 1.054 |
| **35** | 0.841 | 0.990 | 0.720 | 0.685 | 0.965 | 1.002 | 0.949 | 1.005 | 0.995 | 0.969 | 0.996 | 1.073 |
| **40** | 0.845 | 0.982 | 0.704 | 0.683 | 0.960 | 1.002 | 0.932 | 1.006 | 0.995 | 0.971 | 0.983 | 1.080 |
| **45** | 0.846 | 0.979 | 0.690 | 0.675 | 0.956 | 1.003 | 0.916 | 1.009 | 0.994 | 0.970 | 0.976 | 1.082 |
| **50** | 0.851 | 0.973 | 0.682 | 0.662 | 0.954 | 1.004 | 0.901 | 1.008 | 0.997 | 0.961 | 0.973 | 1.106 |
| **55** | 0.858 | 0.971 | 0.672 | 0.653 | 0.953 | 1.004 | 0.885 | 1.009 | 0.998 | 0.963 | 0.965 | 1.118 |
| **60** | 0.861 | 0.972 | 0.663 | 0.646 | 0.954 | 1.005 | 0.871 | 1.014 | 0.995 | 0.965 | 0.961 | 1.120 |
| **65** | 0.866 | 0.972 | 0.657 | 0.637 | 0.958 | 1.006 | 0.857 | 1.013 | 0.998 | 0.963 | 0.958 | 1.145 |
| **70** | 0.873 | 0.973 | 0.650 | 0.628 | 0.963 | 1.006 | 0.844 | 1.015 | 0.999 | 0.965 | 0.955 | 1.160 |

**Table 1:** ratio between the nuclide concentrations obtained with the two codes as a function of the burnup.

# 5. Impact of other parameters on the results

## 5.1. Boron concentration

Simulations with ALEPH2.2 considered cases with water without boron and with a boron concentration of 630 ppm (the same as the ORIGEN-ARP cases). In this section the impact of boron on the neutron emissions is discussed.
The neutron emissions as a function of the burnup obtained with and without boron follow the same trend of Figure 1. The trend of the curves is very similar, but the neutron emissions are higher in the case of presence of boron. The magnitude of the boron impact is reported in Figure 8 where the ratio between corresponding values of burnup is shown. By looking at the ratio, one can estimate that the influence of boron is within 10% of the total neutron emission.

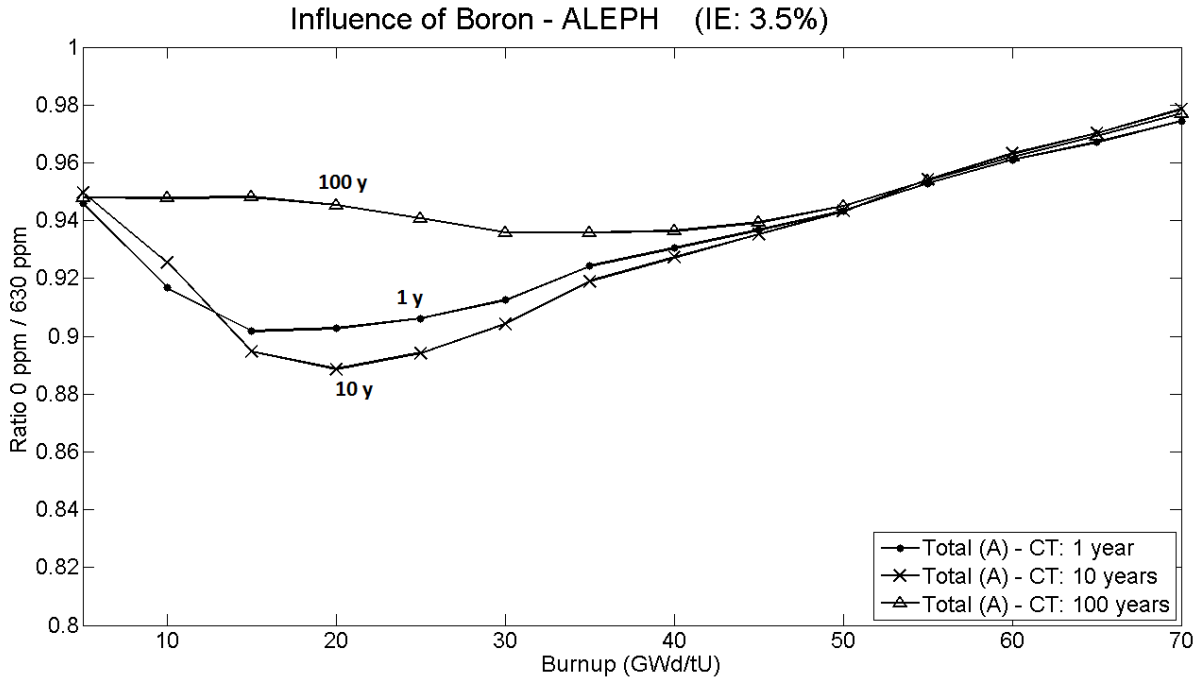



**Figure 8:** influence of the boron concentration as a function of the burnup – ratio of the neutron emission.

Nevertheless the presence of boron not only alters the magnitude of the neutron emission, it also modifies the energy spectrum of the neutron flux. This is because boron is an absorber of thermal neutrons and therefore the spectrum shifts to higher energies (hardened spectrum).
This induces an increase of the resonance captures and a higher production of actinides. Considering that these isotopes are the main neutron emitters, this is a possible explanation of the higher neutron emission observed in the simulations with 630 ppm of boron.

## 5.2. Water spacing between fuel assemblies

One of the relevant information about the geometry of the fuel assembly is the distance between two neighbouring fuel assemblies during the irradiation in the reactor. This parameter depends on the type of fuel assembly and likely also on the particular configuration of the reactor (e.g. number of fuel assemblies in the core).
A set of simulations have been run to quantify the impact of the spacing between fuel assemblies on the concentration of the nuclides and consequently on the neutron and gamma emissions from the spent fuel. The only parameter changing in the simulations was the water spacing between the fuel assemblies (from 0 up to 5 mm). Other characteristics were:

- The initial enrichment of the fuel was 3.5%
- The discharge burnups were 10, 35, and 60 GWd/tU
- The water in the reactor had a constant boron concentration of 630 ppm.

Figure 9 shows the total neutron emission as a function of the water spacing and discharge burnup. The ratio has been calculated normalizing all values to the neutron flux without water gap between the assemblies. From the plot it is evident that the water around the assembly influences significantly the neutron emissions. Already with a water gap of 4 mm (i.e. 2 mm for each side) the total emission is altered by 5% However, this difference decreases with increasing burnup.

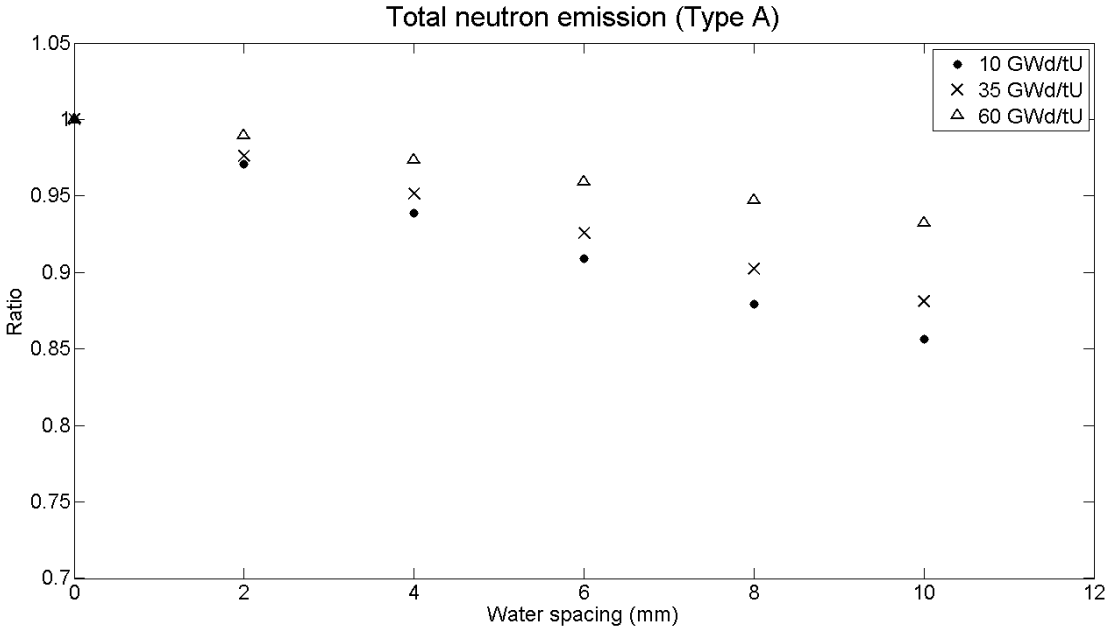



**Figure 9:** water spacing between fuel assemblies: total neutron emission.

No water gap has been placed in the default simulations and this is motivated by the fact that operators try to keep uniform conditions over the whole core cross section. Therefore the pitch between outer rows of neighbouring fuel assemblies should be similar (if not equal) to the pitch of the rods within one single assembly.

## 5.3. Nuclear data library used in the codes

Another important parameter in the simulations is the set of data library that is used. The data libraries available in ALEPH2.2 are the ENDF/B-VII.1 [10] and JEFF-3.1.2 [11] libraries. Both libraries contain full data sets which includes neutron transport data (cross sections and secondary particle emission data), radioactive decay data (which are used by the depletion module for the neutron source calculation), and fission product yields data (which are also used by the depletion module). The choice of the library influences the results because of the different values for the evaluated nuclear data (e.g. nuclides cross sections) adopted in each data set.
Figure 10 shows the ratio of the neutron emissions at discharge calculated using the different libraries as a function of the burnup. The difference in the results is within 10%, with the simulations using the ENDF data library always overestimating the neutron emission compared to the calculations with the JEFF library.
ALEPH2.2 calculates the spontaneous fission neutron source in two ways: first generates the Watt fission spectrum of the neutrons for nuclides undergoing spontaneous fission, with the parameters taken from ORIGEN-S [9] ('Type A' curve in Figure 10). The second one calculates the source using the information stored in radioactive decay data library (ENDF/B or JEFF, 'Type B' curve in Figure 10). Usually, two methods give close results, but a trend is observed for high burnup with the 'Type B' calculations. The trend observed with the calculations arises because ALEPH2.2 recognizes a different set of isotopes contributing to the neutron emission from spontaneous fissions depending on the data library that is used in the calculation. The impact of the nuclides that are missing according to some data library (e.g. $^{252}$Cf is not used if the data library is ENDF/B-VII.1) is not negligible and it is the reason of the trend shown in Figure 10.
It is important to mention that ALEPH is still in the development phase and there are interactions with the development team to investigate this issue.

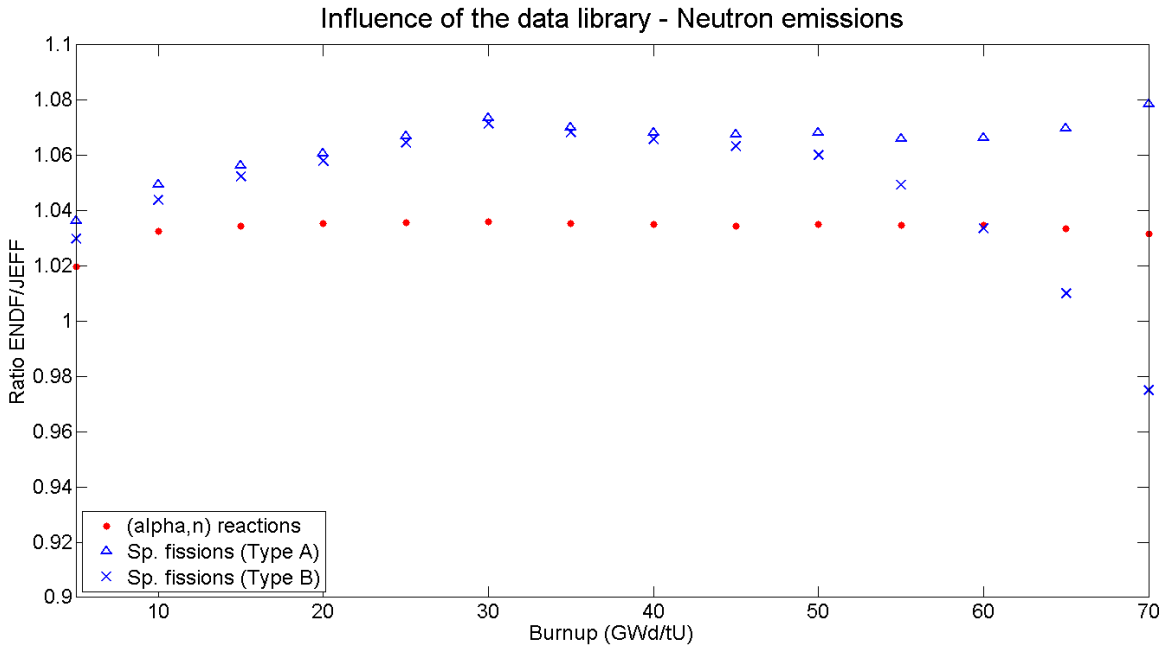



**Figure 10:** influence of the data library: neutron emissions (IE: 3.5% - CT: discharge).

Table 2 shows the ratio between the nuclide concentrations calculated with the two libraries available in ALEPH2.2. The values refer to the fuel with initial enrichment of 3.5% and directly after discharge.
As shown in the comparison between the two codes, also in the case of different data libraries the highest discrepancy is found for curium isotopes ($^{244}$Cm and $^{246}$Cm). The selection of the data library influences most of isotopic concentration within 5% but also for $^{238}$Pu the disagreement is higher and it increases with the burnup.

| Isotope / BU (GWd/tU) | cm242 | cm244 | cm246 | u235 | u238 | pu238 | pu239 | pu240 | pu241 | pu242 | am241 |
|---|---|---|---|---|---|---|---|---|---|---|---|
| **5** | 1.053 | 1.195 | 1.179 | 1.000 | 1.000 | 1.018 | 1.006 | 1.013 | 1.020 | 1.025 | 1.031 |
| **10** | 1.042 | 1.159 | 1.172 | 1.000 | 1.000 | 1.017 | 1.001 | 1.007 | 1.015 | 1.016 | 1.020 |
| **15** | 1.037 | 1.147 | 1.166 | 1.000 | 0.999 | 1.021 | 1.001 | 1.004 | 1.011 | 1.011 | 1.017 |
| **20** | 1.036 | 1.138 | 1.156 | 1.000 | 0.999 | 1.025 | 1.002 | 0.998 | 1.013 | 1.012 | 1.018 |
| **25** | 1.035 | 1.130 | 1.155 | 1.001 | 0.999 | 1.033 | 1.001 | 0.996 | 1.011 | 1.010 | 1.018 |
| **30** | 1.034 | 1.124 | 1.159 | 1.001 | 0.999 | 1.041 | 1.000 | 0.995 | 1.008 | 1.010 | 1.017 |
| **35** | 1.032 | 1.108 | 1.156 | 1.001 | 0.998 | 1.050 | 0.999 | 0.993 | 1.007 | 1.010 | 1.016 |
| **40** | 1.031 | 1.097 | 1.154 | 1.001 | 0.998 | 1.059 | 0.998 | 0.991 | 1.008 | 1.009 | 1.015 |
| **45** | 1.029 | 1.089 | 1.153 | 1.001 | 0.998 | 1.069 | 0.998 | 0.988 | 1.006 | 1.008 | 1.015 |
| **50** | 1.029 | 1.084 | 1.145 | 1.002 | 0.997 | 1.080 | 0.998 | 0.985 | 1.009 | 1.007 | 1.015 |
| **55** | 1.028 | 1.074 | 1.138 | 1.002 | 0.997 | 1.092 | 0.997 | 0.983 | 1.007 | 1.008 | 1.015 |
| **60** | 1.027 | 1.066 | 1.134 | 1.001 | 0.997 | 1.104 | 0.994 | 0.985 | 1.000 | 1.008 | 1.012 |
| **65** | 1.026 | 1.058 | 1.128 | 1.000 | 0.996 | 1.117 | 0.992 | 0.984 | 0.997 | 1.007 | 1.010 |
| **70** | 1.023 | 1.051 | 1.123 | 0.999 | 0.996 | 1.130 | 0.990 | 0.983 | 0.996 | 1.005 | 1.006 |

**Table 2:** ratio between the nuclide concentrations obtained with the two libraries (ENDF/JEFF) as a function of the burnup.

## 6. Conclusions

A reference spent fuel library has been built using the software ORIGEN-ARP and ALEPH2.2 to provide some insights of the different isotopes relevant for neutron emissions. A comparison of the two codes has been performed to check their consistency and the reasons of possible discrepancies.
The neutron emission increases with increasing burnup, whereas there is an opposite trend with initial enrichment and cooling time. Apart from very low burnup values, the spontaneous fissions are the main contribution to the total neutron emission, with (alpha,n) reactions accounting for the remaining.

By looking at the role played by individual isotopes, it is clear that the main contributors are the curium isotopes ($^{242}$Cm and $^{244}$Cm are very important up to 100 years of cooling time, while $^{246}$Cm and $^{248}$Cm arise at high burnup). At low burnup and also at high cooling time there are significant contributions from plutonium isotopes (especially $^{240}$Pu and $^{242}$Pu). As a general consideration, there are always less than 10 isotopes that combined are responsible for about 99% of the total neutron emissions. Only for cooling times higher than 10000 years more isotopes have relevant contributions.
The build-up of several actinides (such as the Cm isotopes) explains the trend of the neutron emissions with initial enrichment, burnup, and cooling time of the spent fuel. In fact, decreasing the initial enrichment or increasing the burnup will lead to a higher fluence level, a higher production of actinides and therefore to a higher neutron emission.
Comparing the two codes used in the simulations, the general agreement is rather satisfactory since the total neutron emissions are within 15%. This is due to a different concentration of actinides calculated by the two codes and possibly also to approximations applied to both models.
The boron added to the water during irradiation in the reactor core is also playing a role, although reduced compared to the main variables associated to the irradiation of the spent fuel (IE, BU, CT). Increasing the boron concentration there will be more production of actinides due to resonant captures and finally higher neutron emissions. Its impact on the total neutron emission can be estimated within 10%.
Other important parameters that affect the calculated composition of the spent fuel are the water spacing between neighbouring fuel assemblies during the irradiation in the reactor (decrease of the total neutron emission of roughly 1% for each mm of gap) and the nuclear data library that is used in the calculations (10% on the total neutron emission).

## Acknowledgements

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**Gamma spectrometry as an early and rapid tool in nuclear forensics**

J.A. Corcho Alvarado; Christoph Wirz ; François Byrde; Béatrice Balsiger; Mario Burger

*Spiez Laboratory; Federal Office for Civil Protection FOCP, 3700 Spiez, Switzerland*

Here we present our gamma spectrometry capabilities for a rapid characterization of nuclear materials. Gamma spectrometry is a very useful technique for screening nuclear materials as most relevant fission and activation products emit gamma radiation. This measurement method is fully non-destructive which enables further investigations of the original samples. High resolution gamma spectrometry was established as our initial tool for determining and characterizing the type of investigated material (e.g. low or highly enriched uranium, plutonium, irradiated materials, etc.). We discuss how gamma spectrometry can be used, for example, to characterize highly enriched uranium (HEU) or plutonium samples. These radionuclides are typically found in illicit trafficking of nuclear materials. In addition, the identification of reprocessed uranium ($^{232}$U) is discussed. This first information is very useful for selecting the steps for further characterization of the material like for example by ICP-MS.

*Keywords: gamma spectrometry, uranium, plutonium, nuclear materials, illicit trafficking*

# Investigation on spectroscopic capabilities in plastic-based Radiation Portal Monitors

**Francesca Rosas[1, 2], Paolo Peerani[1], G. Janssens-Maenhout [2], P. Richir[1]**
1) European Commission, Joint Research Centre, ITU – Ispra (VA), Italy
2) University of Gent, Faculteit Ingenieurs wetenschappen – Gent, Belgium

Institute for Transuranium Elements
Joint Research Centre, European Commission
Via Fermi, Ispra 21020 (VA) Italy
E-mail: francesca.rosas@jrc.ec.europa.eu

***Abstract:***

Radiation Portal Monitors (RPM) based on polyvinyl toluene (PVT) scintillators are the most common gamma ray detectors used in border monitoring. PVT based technology has a poor energy resolution and as such a poor capability for radiation source characterisation, while it provides a high detection efficiency for gamma radiation.

The Energy Windowing analysis can be used to provide reasonable characterization of radioactive material, however this kind of analysis leads to a high rate of innocent alarms due to the presence of Naturally Occurring Radioactive Materials (NORM) in transported goods. The objective of this research is to reduce the innocent alarms on the RPMs that are already deployed worldwide through an improved analysis of the measured signal. The study is being conducted following two different approaches to investigate the spectroscopic capabilities of PVT based RPMs: the Spectra Deconvolution with Monte Carlo simulation and the pattern recognition through Neural Networks Algorithms (NNA). The large variety of spectra needed for both methods has been partially collected. A first set of spectra including Special Nuclear Material (SNM), industrial and medical isotopes were collected by the RPM installed at the JRC-ITU of Ispra, with the support of US-Department of Energy/Second Line of Defense, for the modification of the RPM's acquisition system. A library containing a large variety of spectra of NORM and commonly transported commodities is being collected in-field with an identical RPM, in collaboration with the port of Antwerp (Belgium).

In this paper we will focus on the spectrum unfolding technique based on the inverse response matrix generated with Monte Carlo calculations. The methodology has been validated on synthetic spectra using Monte Carlo simulations. The next step concerns the validation with the analysis of in-field measured spectra. This will allow upgrading the already deployed RPMs.

**Keywords:** nuclear security; photon detection; scintillation detectors; gamma spectrometry; border monitoring.

## *1. Spectra from plastic scintillators*

The typical energy spectrum generated by large-size plastic scintillators exhibits continuum behaviour without any identified peaks [4,5]. Figure 1 compares the spectra from a Cs-137 source acquired using a PVT and a NaI detector. The second one shows a clear and well identified characteristic peak at the photon energy of 662 keV, whereas the first one shows simply a continuum with a slightly pronounced

shoulder (corresponding to the Compton edge at approximately 450 keV). For these reason the classical gamma spectra analysis algorithms, mostly based on peak search methods, cannot be used with spectra from plastic scintillators.

Nevertheless limited spectral information can be derived from PVT detectors. Indeed figure 2 shows a comparison among spectra from different isotopes acquired with a PVT (simulated data). The spectra look different even though there are not identified peaks.

Therefore there is the possibility to profit of these different spectral characteristics.

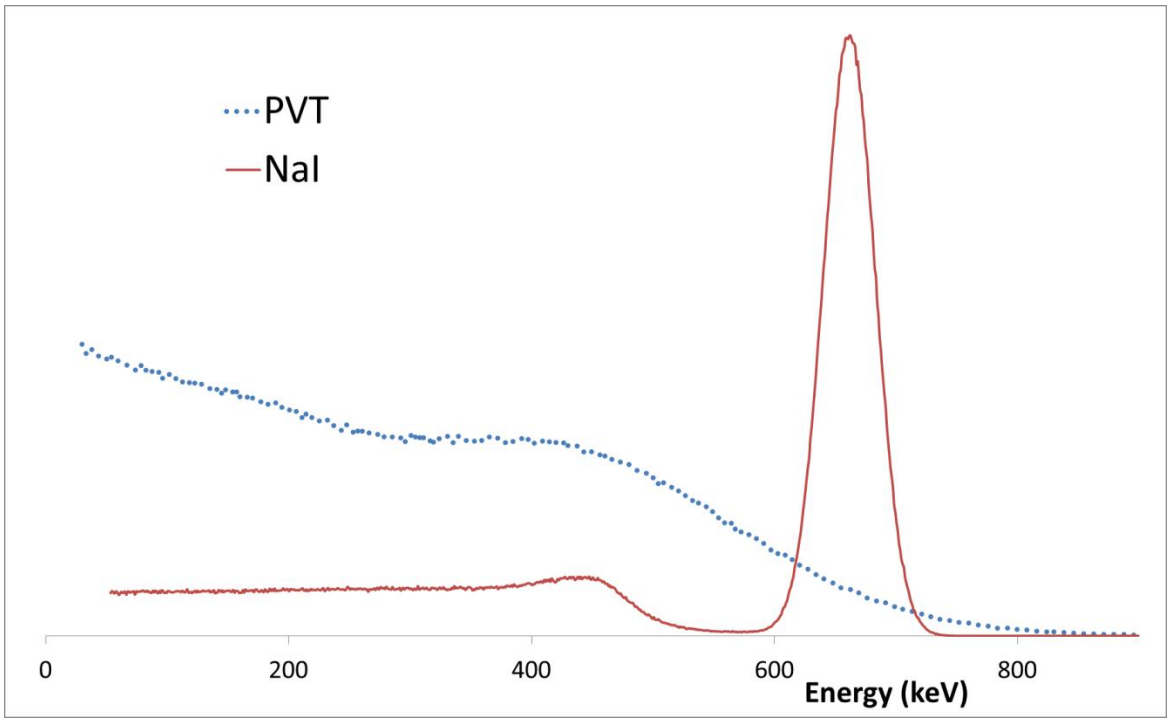



*Fig. 1 – Comparison between Cs-137 spectra measured with PVT and NaI detectors*

**2. Spectrum unfolding algorithm**

Spectrum unfolding is quite a common technique in nuclear physics. The basic idea is to get the response function of a specific instrument to a series of known physical parameters. Then applying the inverse response function to a measurement produced by an unknown input, it should be possible to reach back to the value of the input parameter.

In the case of a gamma spectrometer the spectrum measured by the detector when entered by an incoming photon flux can be generally expressed as:

$$D\ E\ =\ R(E' \rightarrow E)\varphi(E')\,dE' \quad (1)$$

where D(E) is the measured spectrum, φ(E') is the incoming photon flux as a function of energy and R is the response function (probability that a photon with energy E' is detected and measured as having energy E). In general spectra are acquired through multichannel analysers (MCA), so the measured spectrum is not continuous, but a histogram:

$$D\ E_k\ =\ \sum_{j=1}^{N} R(E_j \rightarrow E_k)\varphi(E_j) \quad (2)$$

The response matrix R can be easily obtained when exposing the detector to a series of monochromatic fluxes. For each mono-energetic input, the detected spectrum will correspond to a line of the response matrix:

$$R\ E_j \rightarrow E_k\ = D\ E_k \qquad \text{for} \qquad \varphi\ E\ = \delta(E - E_j) \qquad (3)$$

Once the response matrix is known, the (unknown) input flux that has generated a measured output spectrum can be obtained by solving the equation system (2); in matrix terms, this corresponds to applying to the unknown spectrum vector the transposed inverted response matrix:

$$\varphi\ E_j\ = R_{k,j}^{T}\ ^{-1} D(E_k) \qquad (4)$$

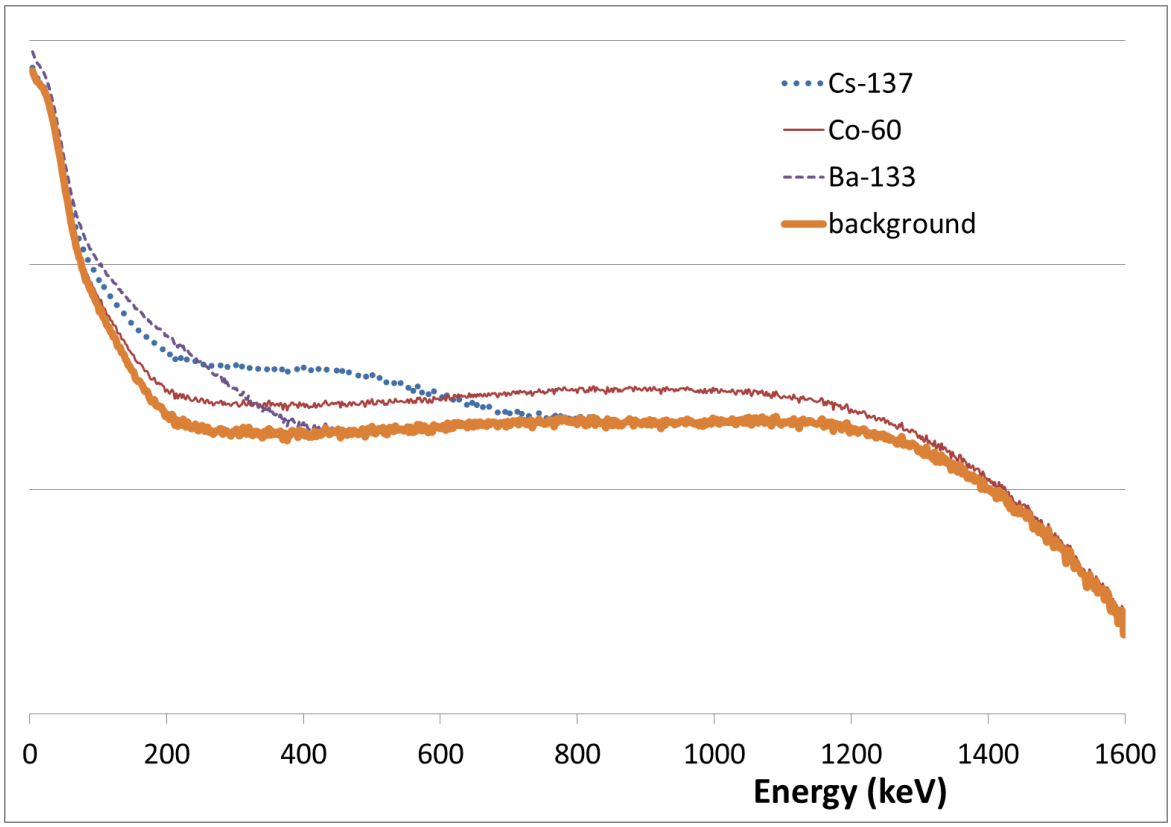



*Fig. 2 – Comparison among spectra from different sources measured with a PVT detector*

Experimental determination of the response function is theoretically possible, but certainly cumbersome. Monte Carlo codes can be used to compute the response of a detector to radiation, so the response matrix can be computed easily by running a series of simulations for a set of N monochromatic sources.

JRC has developed a simple software, called XPortal, to generate the detector response matrix by Monte Carlo technique, based on MCNP [6]. The software allows the user to define the characteristics of the plastic detector (size, composition, density and resolution) and the energy grid (number of groups and bin width). Then it runs automatically a sequence of N Monte Carlo runs; for each run it stores the detected spectrum, corresponding to a row of the response matrix; at the end of the sequence the response matrix is saved together with its transposed inverted matrix. The software then allows deconvolution of any (unknown) spectrum given in input by multiplying it for the inverse response matrix. The spectrum can be either a simulated or a measured one.

### 3. Validation of the method using simulated spectra

A first series of tests of the unfolding algorithm has been done using synthetic spectra generated with MCNP.

### *3.1 Analysis of group structure*

In order to investigate the dependence on the group structure, the energy range of interest (up to 3 MeV) has been divided in 64, 128, 256 or 512 bins and spectra have been unfolded using the 4 group structures. Figure 3 shows the results for a Co-60 source using the different group structures. It is evident that with the broader structures, only the average energy of the photons can be placed around 1.2 MeV, whereas the 512 group structure allows already the discrimination of the two peaks at 1.1 and 1.3 MeV. Therefore we choose to continue our work on the basis of a 512-channels structure, corresponding to a bin width of approximately 6 keV.

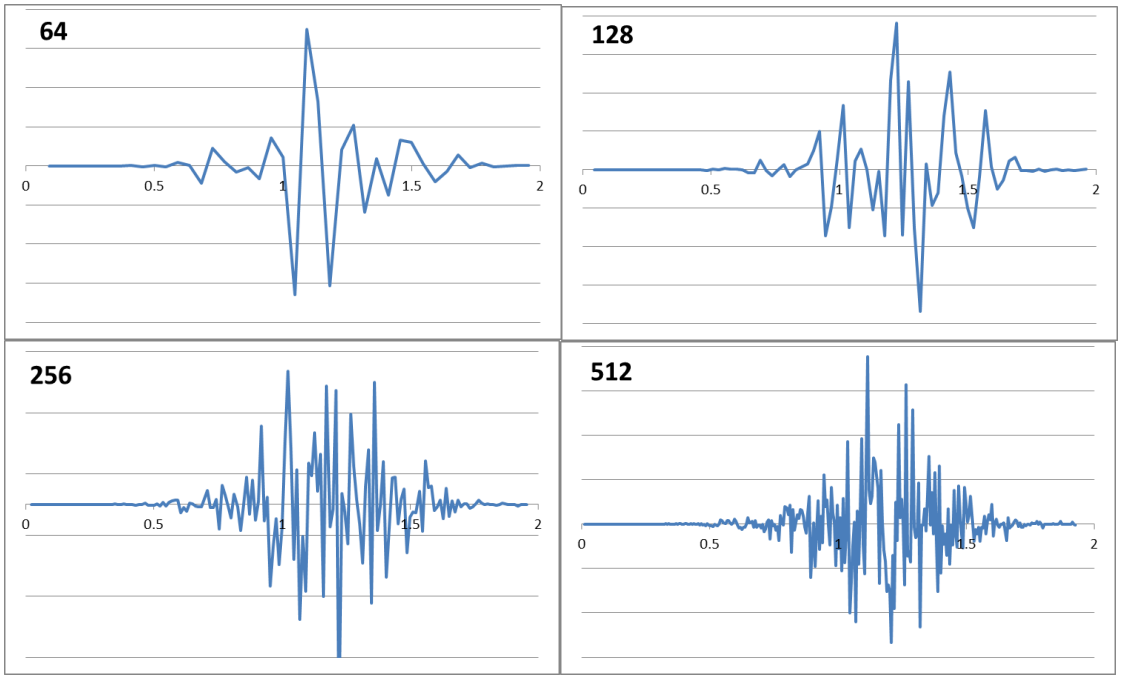


Figure 3 – Comparison of unfolded spectra for Co-60 at different group structures

### *3.2 Fitting raw data*

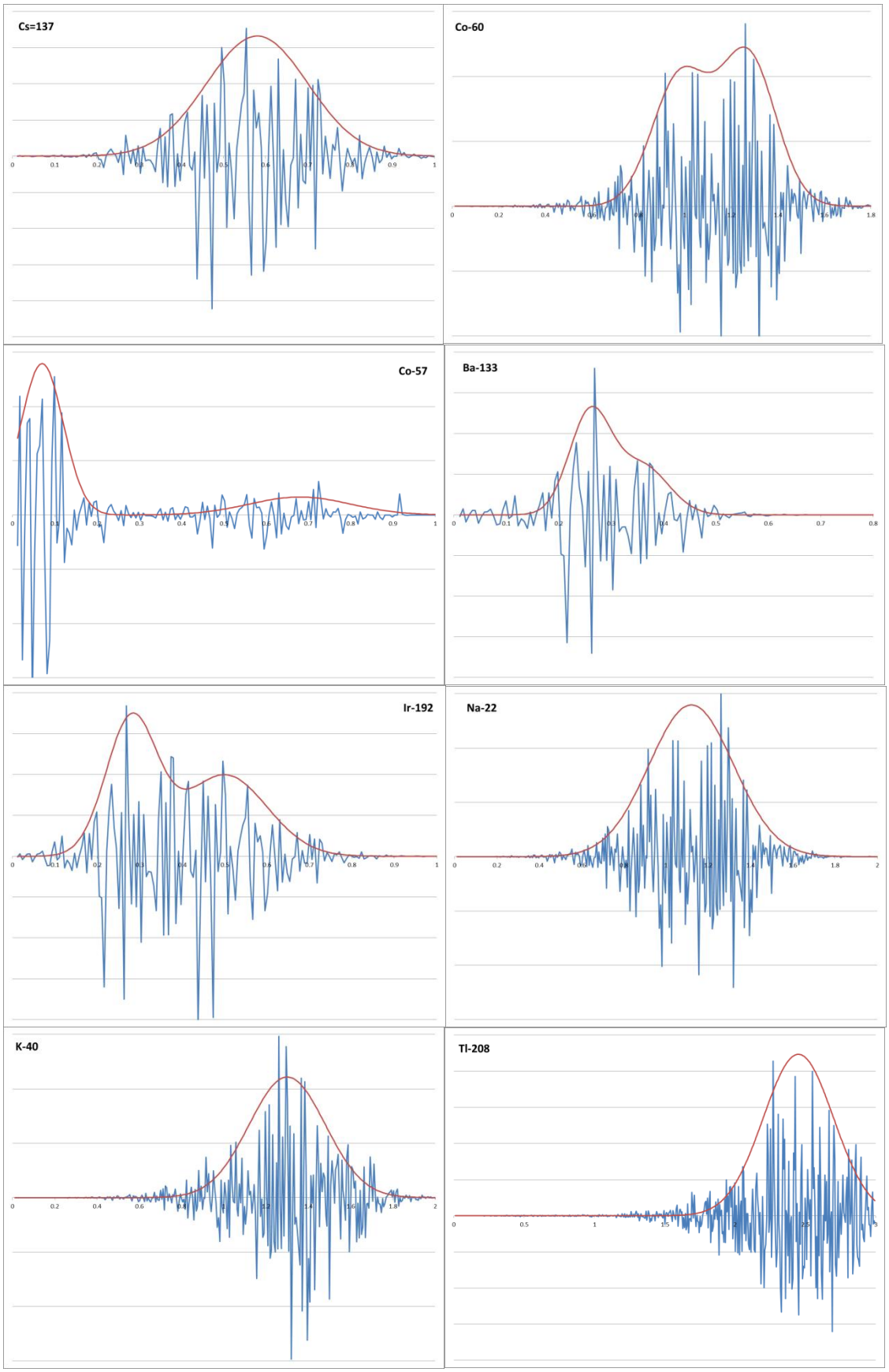
As we can see clearly from figures 3 the data from unfolding are quite noisy. We therefore decided to try a fitting process in order to give them a more classical spectrum-like appearance. The obvious choice was to try a least-square fit using a Gaussian equation:

$$f\ E\ = C * e^{-\frac{(E-E_C)^2}{2\sigma^2}} \qquad (5)$$ where C is a normalisation factor;

$$C = \frac{1}{\sigma\ \sqrt{2\pi}} \qquad (6)$$

and $E_c$ and σ are the fitting parameters representing respectively the Gaussian centroid (corresponding to the energy peak) and the square root of the variance (connected to the "resolution" through the relation: FWHM = 2.355*σ).

Figure 4 shows the raw data resulting from the unfolding of computed spectra from a series of common gamma emitters and the Gaussian fitting.



Figures 4 – Unfolded spectra from several simulated common gamma sources

| Source | Photon energy (MeV) | Centroids (MeV) | σ (MeV) | FWHM (MeV) |
|---|---|---|---|---|
| Cs-137 | 0.66 | 0.58 | 0.12 | 0.28 |
| Co-60 | 1.17<br>1.33 | 0.98<br>1.27 | 0.12<br>0.12 | 0.28<br>0.28 |
| Co-57 | 0.12+0.14<br>0.69 | 0.07<br>0.68 | 0.05<br>0.12 | 0.12<br>0.28 |
| Ba-133 | 0.35+0.38<br>0.30+0.28 | 0.36<br>0.26 | 0.05<br>0.04 | 0.10<br>0.12 |
| Ir-192 | 0.32+0.31+0.30<br>0.47 | 0.28<br>0.50 | 0.06<br>0.10 | 0.16<br>0.24 |
| Na-22 | 1.27 | 1.12 | 0.20 | 0.47 |
| K-40 | 1.46 | 1.30 | 0.18 | 0.42 |
| Tl-208 | 2.61 | 2.45 | 0.25 | 0.58 |

Table 3 – Parameters of the unfolded simulated spectra

**Conclusions**

Table 3 reports the parameters of the fitting Gaussian curves and compares the centroids of the distributions with the energies of the emitted photons for the different sources. From these data we can derive some preliminary conclusions about the performances of the unfolding method:

- The energy range of the photon emission is always well identified
- The centroids of the unfolded peaks are reasonably close to the energy of the emitted photons, always within the 1-σ uncertainties (one standard deviation)
- There is a slight systematic tendency to underestimate the photon energy (the shift increasing at higher photon energies)
- The “resolution” of the unfolding decreases with increasing energy; roughly with a similar trend in most of detectors where FWHM ≈ √E.

The method has been validated through measured spectra of industrial and special nuclear material sources at the JRC of Ispra, using a single PVT scintillator. The spectra were collected through a 512 Channels MCA.
Now we are collecting spectra through the Continuous High Resolution output system for PVT Readout (CHRPR) designed and built at the Pacific Northwest National Laboratories, installed inside our RPM.
Since the CHRPR allows collecting data without interfering with the operation of the RPM, as soon as it will pass the tests in the JRC-Ispra site, it will be installed at the Seaport of Antwerp to collect a large amount of NORM spectra.

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# The verification and sealing of spent fuel casks for long-term dry storage in the EU – Euratom's approach and implementation experience

**A. Kavka, W. Kahnmeyer, P. Schwalbach, P. de Baere, S. Vaccaro**

European Commission – Directorate-General for Energy
Directorate E – Nuclear Safeguards, L-2920 Luxembourg
E-mail: alexander.kavka@ec.europa.eu, wolfgang.kahnmeyer@ec.europa.eu

***Abstract:***

*Spent fuel that is loaded into casks for long-term dry storage becomes inaccessible for credible verification for extended periods of time. It is therefore necessary to acquire the best possible knowledge on the contents of loaded casks and to keep this knowledge in a reliable way. As a general principle, Euratom inspectors verify all spent fuel by the best available methods when it is loaded into casks. NDA measurement of spent-fuel assemblies with an "Ion Fork" gamma-neutron detector has proved to be a useful tool in the Commission's safeguards approach for this purpose. Commission inspectors quantitatively check the data for consistency with the fuel type and irradiation history declared by the power-plant operators. An improved evaluation method, which will be applied in the near future, will be presented. Together with optical surveillance, the fork detector forms an unattended verification system, obviating the need for inspector presence during cask loading. Continuity of knowledge of the cask contents is then kept by using various combinations of seals and surveillance.*

*The methods used for verification and for keeping continuity of knowledge, as well as the related common arrangements with the IAEA, are described for the various types of spent fuel casks that are used in the EU. Possibilities for future improvements of the approach are illustrated.*

***Keywords:*** spent fuel; dry storage; measurement

## 1. Introduction

Safeguards inspection activities related to the dry storage of spent fuel from power reactors require a considerable amount of inspection effort. To some extent they vary due to the types of storage casks used and the operational conditions in the related reactor and storage facilities.

Once stored in a closed cask, the spent fuel items are practically inaccessible for further verification for long periods of time, probably several decades. Keeping in mind that the nuclear material inventory of such casks normally is in the order of 10 – 15 significant quantities of plutonium (1 SQ = 8 kg of Pu), it is obvious that the safeguards authorities need to have a high degree of assurance on the casks contents (amounts of nuclear material) and that they need to keep continuity of this knowledge in a reliable way.

## 2. The Euratom approach for spent fuel cask verification and sealing

From the perspective of the safeguards activities, the loading of spent fuel assemblies into casks for dry storage can be split into two phases, the verification of the contents of the loaded cask and the arrangements to keep this knowledge from the time the cask is being closed.

The Euratom basic requirement for the verification of the contents of casks that becomes practically inaccessible for further verification for long periods of time is to do this verification by best available method and with a high detection probability. Once the material is loaded into a spent fuel cask and the cask is closed there is currently no measurement technique available that could be used to establish the cask inventory with a comparable accuracy [1]. Another reason for measurement during cask loading is that it cannot be fully excluded that some types of casks might be used for final storage without being opened again.

The European Commission (EC) as well as the IAEA currently consider the Ion Fork instrument (FDET) as the best available NDA instrument for the verification of spent power reactor fuel. It is equipped with pairs of neutron and gamma detectors whose output data (gross gamma and neutron intensities) can be used to attain specific information on the fuel assembly.

In general, the EC foresees the use of the Ion fork instrument for all spent power reactor fuel that is loaded into casks for long-term dry storage. The EC requirement is to some extent stricter than current IAEA criteria which foresee such measurements only for fuel where the IAEA considers the fuel pins to be dismountable from the assemblies. The decision whether spent fuel assemblies can be dismantled is however open to interpretation because it may depend on very small technical details of the design of an assembly and is certainly depending on the technology that is or can potentially be used by a nuclear facility operator.

For keeping continuity of knowledge for the contents of spent fuel casks during long-term storage the EC generally uses two independent sealing systems. In most cases this is supplemented by surveillance which is helpful especially when seals need to be removed for operator maintenance purposes on outer cask lids (like checking pressure indicators). In this respect, the EC requirements are fully compatible with the requirements of the IAEA.

The implementation of the above approach has to take into account the specific circumstances at each nuclear installation. This is valid especially for the use of seals which depends on cask design. Differences in this respect will be explained in the next section.

To give a complete picture about the EC safeguards activities related to spent fuel for long-term dry storage, three specific cases need to be mentioned because they deviate to some extent from the above approach:

Spent fuel at the two CANDU reactors that are in operation at Cernavoda NPP in Romania is not measured to the standard of the best available method before being loaded into dry storage baskets. CANDU reactors are refuelled on-load, i.e. spent fuel assemblies are continuously discharged and transferred into spent fuel ponds. However, different from LWR reactor fuel assemblies, CANDU spent fuel bundles are continuously kept under surveillance from core discharge until loading into baskets and transfer to dry storage silos. Assurance about the number of fuel bundles discharged and their irradiation status is provided by the use of core discharge monitors and bundle counters. While stored in the spent fuel ponds, surveillance is supplemented by sealing of spent bundle stacks using underwater ultrasonic seals. The spent fuel in the dry storage silos is then kept under dual seals (metal and COBRA seals).
To some extent in a similar way, spent RBMK reactor fuel at Ignalina NPP in Lithuania is kept under seals and surveillance while being stored in the spent fuel ponds. There is however, no core discharge monitoring instruments in place. This is justified because the two reactors of this plant are under decommissioning. Part of the spent fuel is already stored in dry storage casks which are sealed (using metal and COBRA seals). The spent fuel that is currently stored in ponds under seals and surveillance will finally also be loaded into casks for long-term dry storage.
At Paks NPP, spent fuel is stored in a vault store where each fuel assembly is put into a separate storage channel. When being transferred into the store, each fuel assembly passes neutron and gamma monitors. They are however not configured in a similar way as an ion fork. So far, the measurement results have not been subject to a similar evaluation as ion fork data. Spent fuel assemblies in the store remain in principle accessible because they can easily be retrieved from their positions which justifies the relaxation of the general principle of measuring them to the best available method. While being stored, groups of spent fuel assemblies are kept under dual sealing (metal and COBRA seals). In addition, surveillance is used mainly to cover the process of transfers of incoming fuel but potentially also as a backup to seals in case a seal layer fails.

## 3. Current implementation

Storage of spent fuel in casks for long-term dry storage is currently only done in installations in the non-nuclear weapon states of the EU where the EC applies safeguards on all nuclear material together with the IAEA[1]. It was therefore important for the EC to agree on common requirements for the verification and sealing of spent fuel for long-term storage in casks. This was done in line with the New Partnership Approach of 1992 which calls for commonly agreed safeguards approaches, inspection activities and use of instruments between IAEA and EC. As a result, a procedure was developed, and agreed in 2009, which describes the common IAEA/EC arrangements during the loading and sealing of spent fuel casks for long-term dry storage. It addresses the safeguards relevant steps during preparation of a loading campaign, cask loading, cask sealing and at the end of a campaign. It is an important element of this procedure that it provides for cask sealing being done by an operator (or an EC inspector) alone which can significantly reduce inspection effort. Such sealing by operators is based on the fulfilment of specific conditions, especially that the cask contents has previously been successfully verified by the EC and IAEA inspectors, and that assurance can be achieved about the sealed object and its seal. This assurance is done, first, through keeping continuity of knowledge of the verified cask contents by surveillance cameras until sealing. Then, the sealing is done by using a COBRA seal with an immobilized seals verifier and by using a VACOSS or EOSS (in VACOSS mode) that is connected to a DCM-14 based camera module. As a result, it can be guaranteed that sealing has been done with a dual sealing system inside the containment of the reactor before the cask leaves surveillance. The COBRA and VACOSS/EOSS seals are then usually verified at the latest during the subsequent PIV inspection in the dry store, or after the end of the loading campaign. All measurement, surveillance and seals data, as well as operator declarations are shared between EC and IAEA. For those cases where the IAEA does not require measurement of fuel assemblies by FDET (see section 2) the EC will provide their measurement data to the IAEA. In a summary, the agreed approach reflects the IAEA integrated safeguards criteria and allows both EC and IAEA to fulfil their respective requirements.

Data acquisition in the FDET measurement system normally runs for weeks or even months, for the entire duration of the spent-fuel shipment campaign. Data are stored at an interval of typically 10 seconds. To measure the gamma and neutron emission rates from a spent-fuel assembly, the assembly is inserted between the arms of the fork detector for about 2 minutes, so that several data points are obtained with the assembly fully inside the detector.

The two most frequently used detector models have dimensions adapted to the typical size of a PWR and BWR fuel assembly, with a 24- or 17-cm separation between the fork arms, respectively [2].

The data-acquisition electronics and computer are in a sealed cabinet. An array of lights, visible from the outside, confirms to the operator that the system is working. Thus the system can run in unattended mode. Inspectors can download the measured data when each cask has been loaded or when the whole campaign is over.
The presence of gamma and neutron data significantly above background is, in itself, an indication of spent fuel and thus a relevant safeguards result, immediately verifiable on the data-acquisition computer during the inspection.

The following two graphs show such typical gamma and neutron results acquired during the loading process of a CASTOR cask with 19 PWR fuel elements (15 LEU and 4 MOX).

[1] At beginning of 2013, EDF Energy has announced the start of construction of a facility for the dry storage of spent fuel in casks at the Sizewell B power station in the UK. This will be the first such installation where the EC will implement its approach for spent fuel cask verification and sealing in a nuclear–weapon state of the EU.

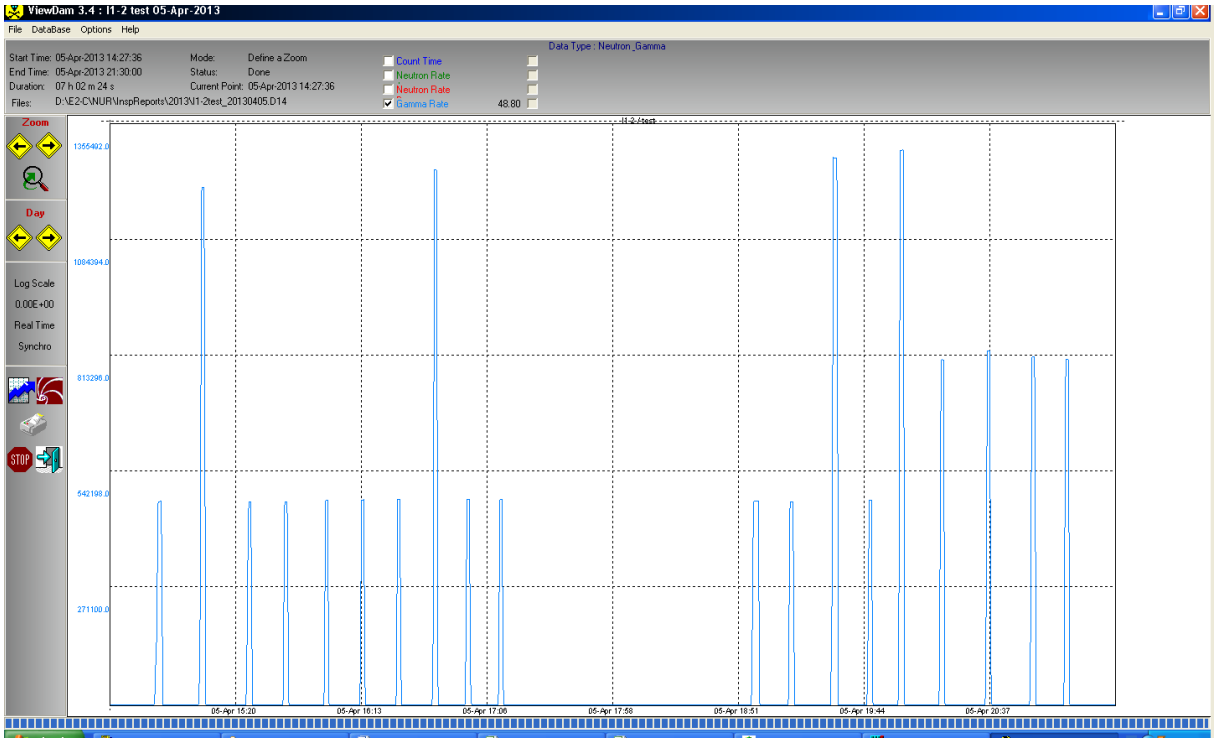


**Figure 1**: Gamma acquisition results (19 PWR fuel assemblies)

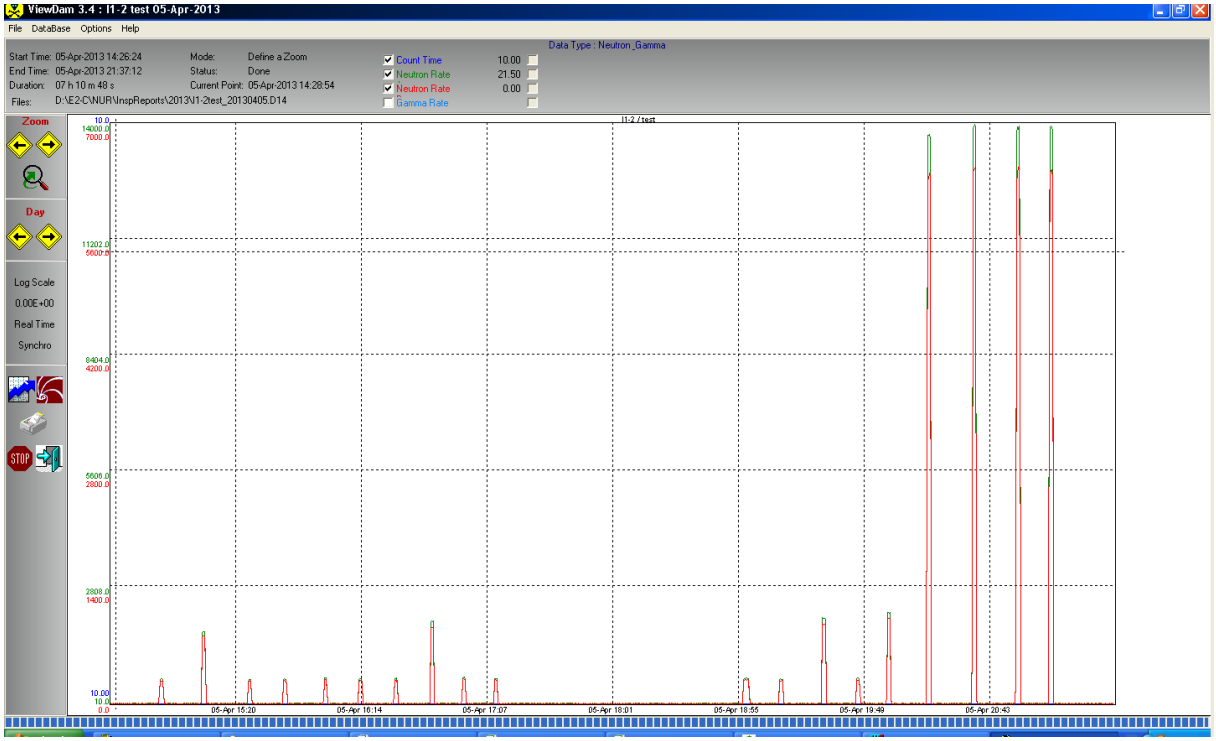


**Figure 2**: Neutron acquisition results (19 PWR assemblies, with 4 MOX assemblies)

After the inspection, a more detailed quantitative evaluation of the data is performed at Euratom headquarters. The purpose of this analysis is to check if the relative gamma and neutron intensities from a set of fuel assemblies are consistent with the fuel characteristics declared by the operator. The data set considered can be the fuel loaded into one or several casks in one campaign, since the efficiency calibration of the Ion Fork system, even if unknown, can be assumed constant throughout the campaign.

The following information is requested from the operator for each fuel assembly: The initial nuclide composition, the irradiation history and the final burnup.

The measured data are analysed on the basis of the following simplified physical model [3]:

- When a fuel assembly is discharged from the reactor, the amount of fission products is proportional to the burnup. As the fission products decay, the gamma-ray intensity decreases over time. For all assemblies, the gamma intensity divided by burnup is the same function of cooling time. After several years, the short-lived nuclides are largely gone and the gamma rays are mostly due to $^{137}$Cs (half life 30 years) and $^{134}$Cs (2 years). The gamma intensity from this combined source can be described by a function proportional to $T^a$, where T is the cooling time and the exponent *a* is between –0.4 and –0.7.

- The neutron emission after about 3 years' cooling is mainly due to spontaneous fission of $^{244}$Cm (half life 18 years). The amount of $^{244}$Cm at discharge is an increasing function of the burnup, proportional to $B^b$, where B is the burnup and the exponent *b* depends on the type of fuel, i.e. its initial $^{235}$U enrichment or whether it was initially MOX. Typical values of *b* are around 4.

The measured gamma and neutron intensities are displayed in two graphs:

- gamma count rate (G) divided by burnup (B), vs. cooling time (T),
- neutron count rate (N) vs. burnup (B)

in which the curves $G/B = c\ T^a$ and $N = k\ B^b$ are least-squares-fitted, using for example Microsoft Excel. In most cases we have obtained a good fit, with parameters *a* and *b* close to the model values, indicating consistency with the operator's declarations.

Since 2009, only looking at installations in Germany alone, Euratom inspectors have measured and analysed in this way spent PWR and BWR fuel loaded into 126 CASTOR casks.

Our experience has shown some recurring features of the Ion Fork data:

- MOX fuel is clearly recognised by the neutron intensity, which is often an order of magnitude higher than for $UO_2$ fuel at similar burn-up and cooling time. (A similar but much less dramatic effect exists for $UO_2$ fuel assemblies with different initial $^{235}$U enrichment: The neutron emission is expected to decrease with increasing enrichment, at the same burn-up and cooling time. Thus, a separate curve could be fitted for each initial enrichment but in most cases the differences in initial enrichment are too small to make the effect visible.)
- A gamma intensity lower than expected is often due to a fuel assembly that was irradiated, then cooled for one or more cycles, then used again in the reactor. If, as a rough approximation, the partial cooling times are simply added to a total "effective" cooling time, the data from different assemblies often correlate well.
- Irregularly scattered data points may suggest misidentified fuel assemblies, perhaps due to an error in the time sequence of cask loading declared by the operator.

Typical results of the above qualitative evaluation of FDET measurement results are shown in the following graphs.

Figure 3 shows the evaluation of gamma emission data, as described above, for 19 PWR elements based on the cooling time after final discharge. Figure 4 shows this data when taking into account an effective cooling time (affecting 3 elements having a much longer effective cooling time if compared to their cooling time after final discharge).

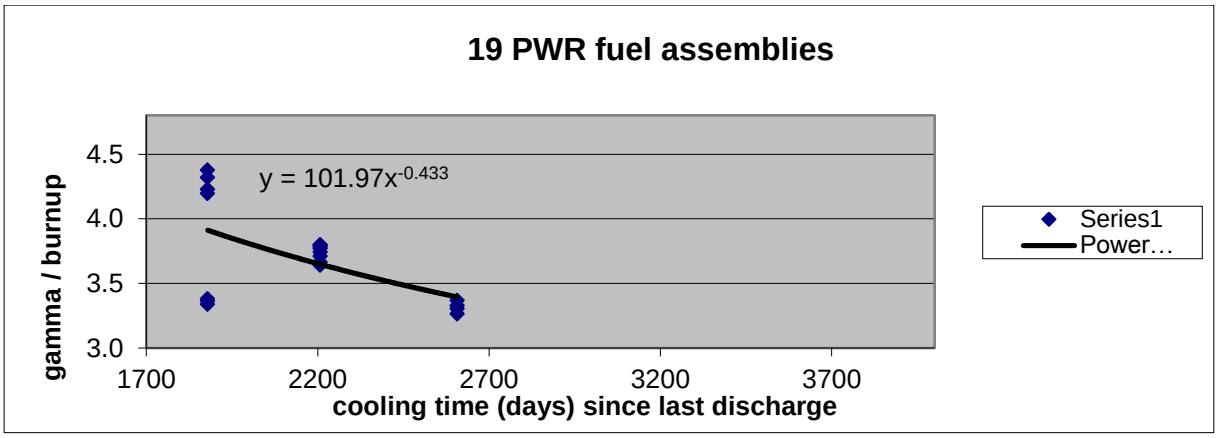


**Figure 3**: G/B versus cooling time after final discharge (19 PWR elements)

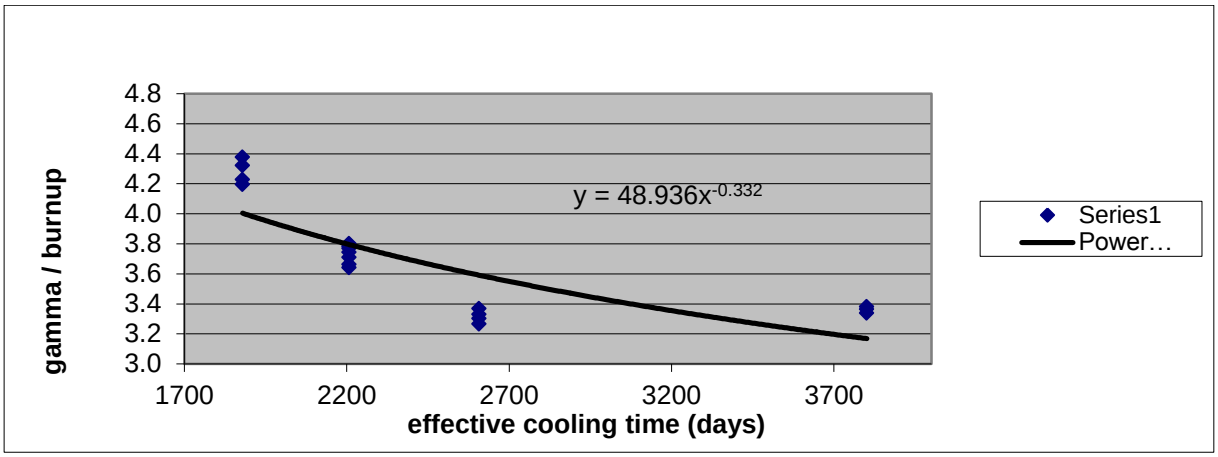


**Figure 4**: G/B versus effective cooling time (19 PWR elements)

In Figures 5 and 6, the evaluation of neutron emission data, as described above, is presented for 19 PWR elements, separately shown for LEU elements (Fig. 5) and MOX elements (Fig. 6) because of the differences in neutron count rates.

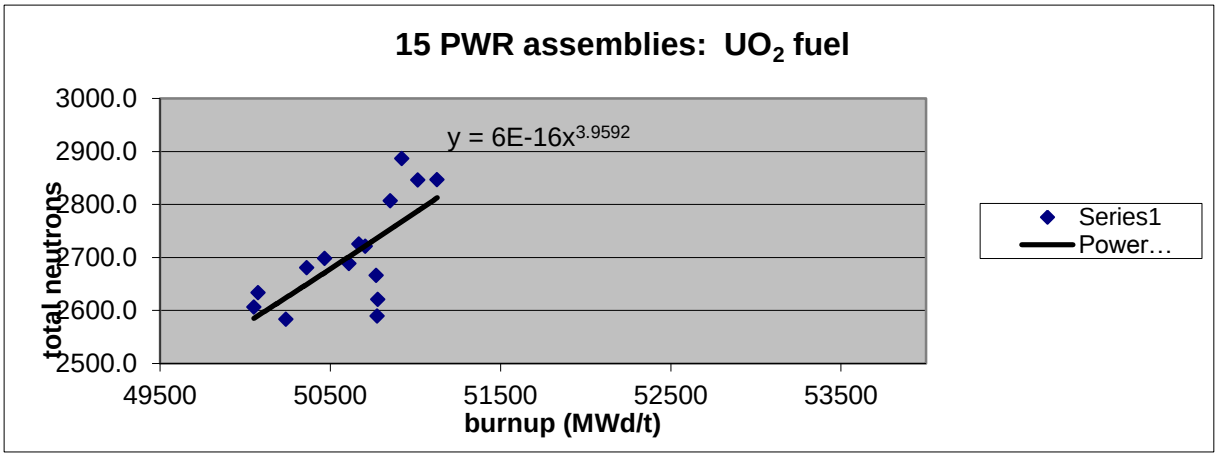


**Figure 5**: Neutron count rate versus burnup (LEU fuel)

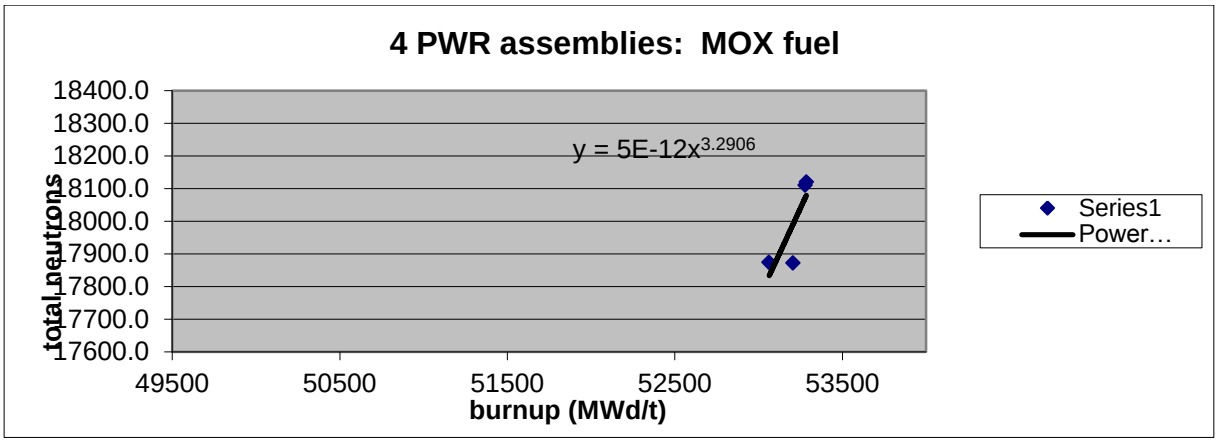


**Figure 6**: Neutron count rate versus burnup (MOX fuel)

In the majority of cases, the sealing of casks can be done using the above procedure while the casks are still in the reactor hall. There are however cask designs where this is not possible. Then, the casks have to be followed from their removal from the reactor hall until their sealing at their place of storage which is a rather time consuming process.

In the spent fuel stores, the casks are usually kept under dual sealing by individual metal and COBRA seals.

## 4. Improvement of the evaluation of Ion fork measurement results

The current evaluation of FDET data has a number of disadvantages because it is fully qualitative. Therefore the EC had started cooperation with Oak Ridge National Laboratory (ORNL) to develop a more sophisticated possibility for the analysis of FDET data based on the use of modelling reactor operation to calculate gamma and neutron emission rates. To allow an immediate evaluation of measurement results during an inspection, modelling based on ORIGEN [4] has been integrated into the EC's CRISP software for the evaluation of data that have been acquired using the EC's standard RADAR data acquisition tool.

To allow evaluation already during the inspection, the irradiation history has to be known to the inspectors most preferably in electronic format. This will then allow to automatically feed operator data into the modelling on the one hand and to compare them with the measurement events that were identified by CRISP.
Currently, arrangements are being made with operators for the provision of irradiation history information in electronic format.

The following graphs show the comparison between measured gamma and neutron emission data calculated using ORIGEN. Although this method [5] will still have to be tested against different types of fuel (including MOX) it can already be stated that the quality of FDET data evaluation by the EC will significantly improve.

Figures 7 and 8 show a comparison between calculated and measured gamma and neutron emission data for 19 PWR fuel elements.

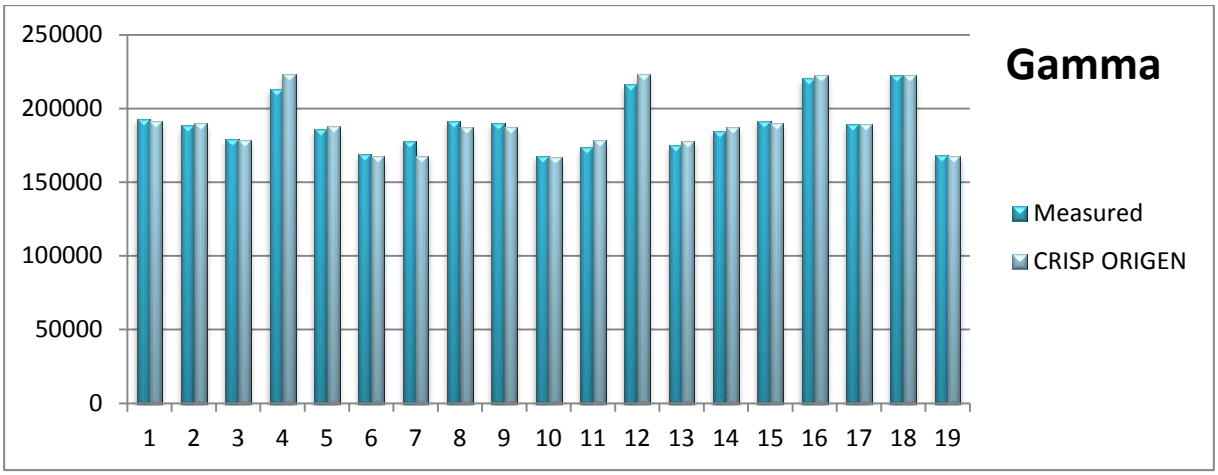


**Figure 7**: Comparison between calculated and measured gamma emission data for 19 PWR fuel elements

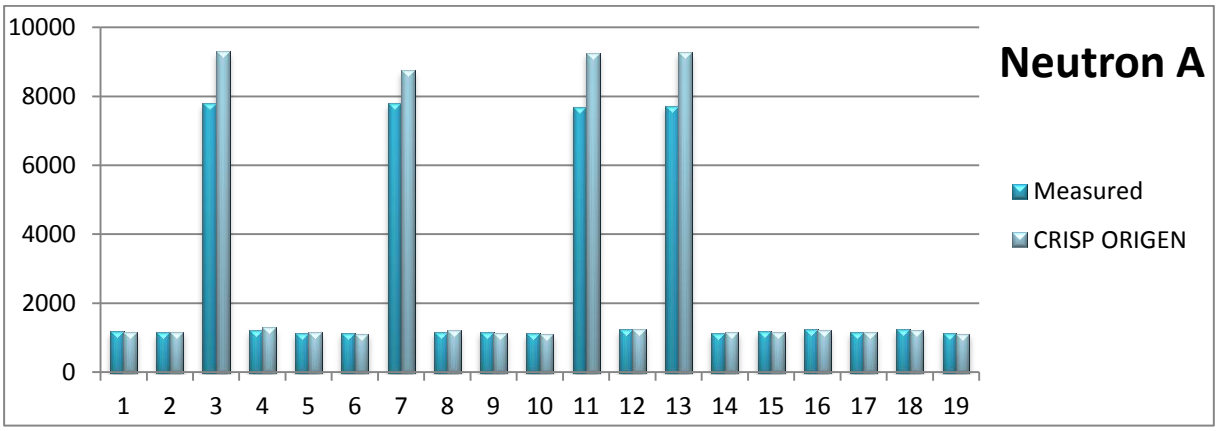


**Figure 8**: Comparison between calculated and measured neutron emission data for 19 PWR fuel elements

In a number of cases, the sealing of casks is not yet done by operators because of a lack of assurance on the side of the operators that seals were correctly applied. With COBRA seals this is solved because the operator can visually observe the COBRA image taken. For EOSS (in VACOSS mode), a specific tool is currently being developed that will allow the operator to get information about the seal status and to store this information in order to have a proof about the correct closing of the seal. Full implementation of this sealing by operator will significantly reduce inspection effort and, at the same time, reduce operator effort for coordination with the inspectorates for the sealing done by inspectors.

## 5. Conclusions

The evaluation of FDET measurement results is currently done in a simple qualitative way which has deficiencies especially when dealing with fuel of non-uniform irradiation history or a mix of LEU and MOX fuel. The introduction of a more reliable evaluation method is being started which will combine electronic input of irradiation history data, reactor modelling using this data and comparison of the modelling results with measured gamma and neutron data.
Future experience from using this method will also be valuable for the development of an automated, fully unattended method for the evaluation of FDET measurements in future encapsulation plants.

Further efficiency gains for the inspectorates are expected through the wider use of cask sealing by operators based on reliable feed-back to operators on the successful application of seals. This especially important in view of the expected increase of spent cask loading activities in the coming years.

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# Monitoring Uranium Mining and Processing Sites Under Decommissioning Using Hyperspectral Imagery

**Irmgard Niemeyer[1], Clemens Listner[1], and Sven Martens[2]**

[1)] Forschungszentrum Jülich GmbH, IEK-6: Nukleare Entsorgung und Reaktorsicherheit, 52425 Juelich, Germany
[2)] RWTH Aachen, 52062 Aachen, Germany
E-mail: i.niemeyer@fz-juelich.de, c.listner@fz-juelich.de, sven.martens@rwth-aachen.de

***Abstract:***

*Previous safeguards-related studies using hyperspectral imagery focused on the possibilities of extracting unique features of uranium mining and milling, and of verifying States' declarations, such as movement of material. If these unique features existed, they could be used for identifying potential clandestine activities as to uranium mining and processing. However, no specific "uranium signature" was found so far.*
*The objective of the given study was to assess the potential of hyperspectral imagery analysis for detecting environmental contaminations due to uranium mining and processing. Though intended for environmental monitoring purposes, the study was supposed to give also some implications on monitoring uranium mining and processing sites from a nuclear safeguards perspective.*
*Hyperspectral data were acquired near Zwickau/Saxony over Wismut uranium mining legacies under rehabilitation in the context of an airborne hyperspectral campaign using the HyMap Imaging Spectrometer. Simultaneously to image acquisition, the surface reflectance was measured in te area of interest using the ASD Fieldspec Pro spectrometer, and soil and vegetation samples were collected and subsequently analyzed by mass spectrometry (ICP-MS).*
*The first project phase was intended to evaluate new hyperspectral data classifiers and change detection methods. The second project phase aimed at integrating hyperspectral remote sensing and the in-situ measurements by applying traditional and new supervised hyperspectral classifiers.*
*The given paper reports on the second project phase and summarizes the overall findings from either phase.*

**Keywords:** uranium mining and processing; hyperspectral imagery; integrated classification

## 1. Introduction

Hyperspectral sensors record the reflected radiation in several hundreds of very narrow contiguous or overlapping wavelength bands from visible to mid infrared. Using well-calibrated hyperspectral data allows to quantitatively estimate geophysical, geochemical and biochemical characteristics of the earth's surface. From a safeguards perspective, hyperspectral data might be useful for assessing surface cover information due to uranium ore drilling, mining and processing activities. By fusing the results of lower resolution hyperspectral analysis results with spatial high resolution imagery, both geometric and materials properties of surface objects could be extracted.

Previous safeguards-related studies focused on the possibilities of extracting unique features of uranium mining and milling, and of verifying States' declarations, such as movement of material [1]. If these unique features do exist, they could be used for identifying potential clandestine activities as to uranium mining and processing. However, no specific "uranium signature" was found so far. In an earlier study, endmember spectra were extracted from airborne SWIR Full Spectrum Imager (SFSI-2), acquired over the Candian processing site Key Lake (Saskatchewan) [2]. The authors concluded that no uranium compounds were identified from the airborne data. Another study used airborne Probe-1

data acquired over the Canadian Pronto mine (Ontario) in order to distinguish uranium mine tailings from other types of mine tailings based on mineral absorption features [3]. Here, the authors finally stated that no spectral absorption feature could be associated with uranium mine tailings.

Today, two satellite-based instruments provide hyperspectral data, the Hyperion Imaging Spectrometer flying onboard NASA's Earth Orbiter-1 (EO-1) spacecraft since 2000 and the Compact High Resolution Imaging Spectrometer (CHRIS) flying onboard ESA's Proba-1 since 2001. Hyperion provides 220 spectral bands in the reflective solar wavelength region from 0.4 to 2.5 µm with contiguous spectral coverage and bandwidths of 10 nm at a scene size of 7.5 km by 100 km [4]. The circular sun-synchronous orbit in 705 km at 98.7° declination allows matching within one minute the Landsat-7 orbit [5]. CHRIS measures directional spectral reflectance in the spectral range of 0.415–1.050 µm at a spectral resolution of 5–12 nm on a swath of 13 km. Using multiple viewing and illumination geometries, image data with 17-20 or 34-40 m spatial resolution in 18 or 62 narrow spectral channels respectively can be acquired [6].

Thus, CHRIS offers the spectral reflectance in the visible and near infrared spectrum only. The spatial resolution of 30-40 m might be a limiting factor for applying hyperspectral data in a number of applications, such as the extraction of safeguards-relevant information. Moreover, the Hyperion image data involves huge noise effects, i.e. the signal to noise ratio hinders the analysis. The latter, however, is expected to be improved by the Canadian Hyperspectral Environment and Resource Observer (HERO) mission [7] and the German Environmental Monitoring and Analysis Program (EnMap) [8].

EnMAP will be a dedicated imaging pushbroom hyperspectral sensor covering the spectral range from 0.430 to 0.950 µm (VNIR) and from 0.950 to 2.400 µm (SWIR) with 184 channels, a swath width of 30 km at high spatial resolution of 30 m, off-nadir (30◦) pointing feature for fast target revisit (<3 days) and an envisaged 4-day revisit. [8]

In preparation of the EnMAP mission, the German Remote Sensing Data Center (DLR/DFD) carried out an airborne hyperspectral campaign in summer 2009 (HyEurope 2009) [9] using the HyMap Imaging Spectrometer [10]. One area of interest was located next to Zwickau/Saxony at the Wismut uranium mining legacies under rehabilitation. Simultaneous to image acquisition, the surface reflectance was measured in the area of interest using the ASD Fieldspec Pro spectrometer, and soil and vegetation samples were collected and subsequently analyzed by mass spectrometry (ICP-MS).
The objective of the study was to assess the potential of hyperspectral imagery analysis for detecting directional or diffuse environmental contaminations due to uranium mining and processing. Though intended for environmental monitoring purposes, the study was supposed to give also some implications on monitoring uranium mining and processing sites from a nuclear safeguards perspective.

The first project phase was intended to evaluate new hyperspectral data classifiers and change detection methods [11]. The second project phase aimed at integrating hyperspectral remote sensing and the in-situ measurements by applying traditional and new supervised hyperspectral classifiers. The given paper reports on the second project phase and summarizes the overall findings from either phase.

## 2. Area of Interest & Data

### 2.1. Area of Interest

Our area of interest was located next to Zwickau/Saxony at the Wismut uranium mining legacies under rehabilitation. The Wismut Company was established in Saxony in 1947, initially run by the Soviet military, later in Soviet and then Soviet-German ownership. Wismut, exploiting the German uranium deposits for the Soviet nuclear program, produced a total of 231,000 tons of uranium during the four decades of its existence. This ranks Wismut as number three in world-wide uranium production after the US and Canada. Following German reunification, uranium mining was terminated in the end of 1990. Since mid 1991, Wismut GmbH has become a German Government-owned company operating in Saxony and Thuringia on decommissioning, cleanup, and rehabilitation of the former uranium mining and processing sites. [12]

We selected the Crossen site as one of the former processing facilities. Today, the location consists of the dismantled processing site (Figure 1a), the tailings dump (Figure 1b) and the tailings ponds Helmsdorf/Daenkritz I (Figure 1c).

## 2.2. HyMAP data

The HyMap™ scanner manufactured by Integrated Spectronics Pty Ltd. provides 128 bands across the reflective solar wavelength region of 0.45 to 2.5 µm with contiguous spectral coverage (except in the atmospheric water vapor bands) and bandwidths between 15 and 20 nm. The Hymap provides a signal to noise ratio (>500:1) and image quality that is setting the industry standard. [10]
During the HyEurope campaign, two images were acquired on August 20, 2009. The first scene was obtained in the afternoon (15:37 local time) with the best possible spatial resolution of 5m (Figure 1), the second was recorded half an hour later (15:52 local time) at a spatial resolution of 9.5m (Figure 2). The images were atmospherically and geometrically corrected by DLR/DFD before delivering to us.


**Figure 1:** Area of interest given by HyMAP on August 20, 2009: Wismut former processing site Crossen with the dismantled processing site (a), the tailings dump (b) and the tailings pond Helmsdorf/Daenkritz I (c). Flight time: 13:37 UTC; flight height: 2718m; ground resolution: 5m.

**Figure 2**: Area of interest given by HyMAP on August 20, 2009. Flight time: 13:52 UTC; flight height: 5058m; ground resolution: 9.5m.

## 2.3. In-situ Measurements

Simultaneous to image acquisition, a field campaign was carried out in order to

- measure the surface reflectance using the Analytical Spectral Devices Inc. (ASD) Fieldspec Pro spectrometer, capable of recording reference spectra in 2151 bands in the 0.35 to 2.5 µm spectral range (Figure 3);
- select soil and vegetation samples and subsequently analyze heavy metal concentrations in the samples by inductively-coupled-plasma mass-spectrometry (ICP-MS);
- map the surface cartographically and measure GPS points.

In addition, we received environmental data from Wismut such as terrestrial surveying, climate and water quality data.


**Figure 3:** Surface reflectance using the ASD Fieldspec Pro spectrometer. Left: Yellow stars indicate the locations of measurements. Right: Two spectra examples.

# 3. Methodologies

The first project phase aimed at evaluating unsupervised hyperspectral data classifiers (class-dependent neural networks) and change detection methods (regularized iteratively reweighted multivariate alteration detection) [11]. The objectives of the second phase were to integrate hyperspectral remote sensing and in-situ measurements (reference spectra), and thus to evaluate supervised hyperspectral data classifiers. For classification, spectral angle mapper and matched filtering were selected. The aim is to identify spectral anomalies and/or significant spectral pattern outside the former site that could indicate impact by the past uranium processing activities.

## 3.1 Spectral Angle Mapper Classification

The Spectral Angle Mapper (SAM) calculates the spectral similarity between a test reflectance spectrum and a reference reflectance spectrum assuming that the data is correctly calibrated to apparent reflectance with dark current and path radiance removed. The spectral similarity between the test (or pixel) spectrum *t* and the reference (or laboratory) spectrum *r* is expressed in terms of the average angle $\Theta$ between the two spectra as calculated for each channel *i* (Equation 1). [x]

SAM treats the spectra as vectors in a space with dimensionality equal to the number of bands *n* (Figure 4). The outcome of the spectra angle mapping for each pixel is an angular difference measured in radians ranging from zero to $\pi/2$ which gives a qualitative estimate of the presence of absorption features which can be related to mineralogy. Smaller angles represent closer matches to the reference spectrum. Pixels further away than the specified maximum angle threshold in radians are not classified. [x]

$$\Theta = \cos^{-1}\left(\frac{\sum_{i=1}^{n} t_i r_i}{\sqrt{\sum_{i=1}^{n} t_i^2 \sum_{i=1}^{n} r_i^2}}\right) \quad (1)$$

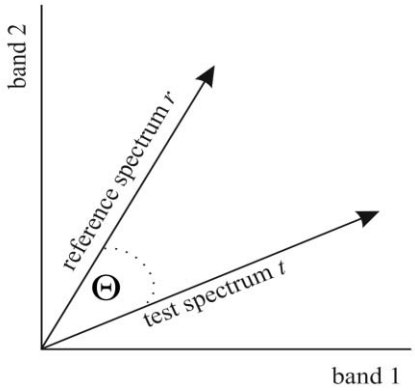


**Figure 4:** Spectral Angle Mapper

### 2.2 Matched Filtering

Matched filtering maximizes the response of a known endmember and suppresses the response of the composite unknown background, thus “matching” the known signature. Matched filtering performs a partial unmixing that finds the abundances of user-defined endmembers. It therefore represents a rapid means of detecting specific minerals based on matches to specific library or image endmember spectrum and does not require knowledge of all the endmembers within an image scene. Matched filter results are presented as grey-scale images with values from 0 to 1.0, estimating the relative degree of match to the reference spectrum (where 1.0 is a perfect match). [14]

## 3. Experiments

### 3.1. Spectral Angle Mapper Classification

The SAM classification was applied using all acquired field spectral reflectances (as reference spectra (Figure 4). The ENVI software GUI enables to switch on/off the results for selective reference spectra, however, a clear visualization of all (> 150) spectra becomes infeasible.

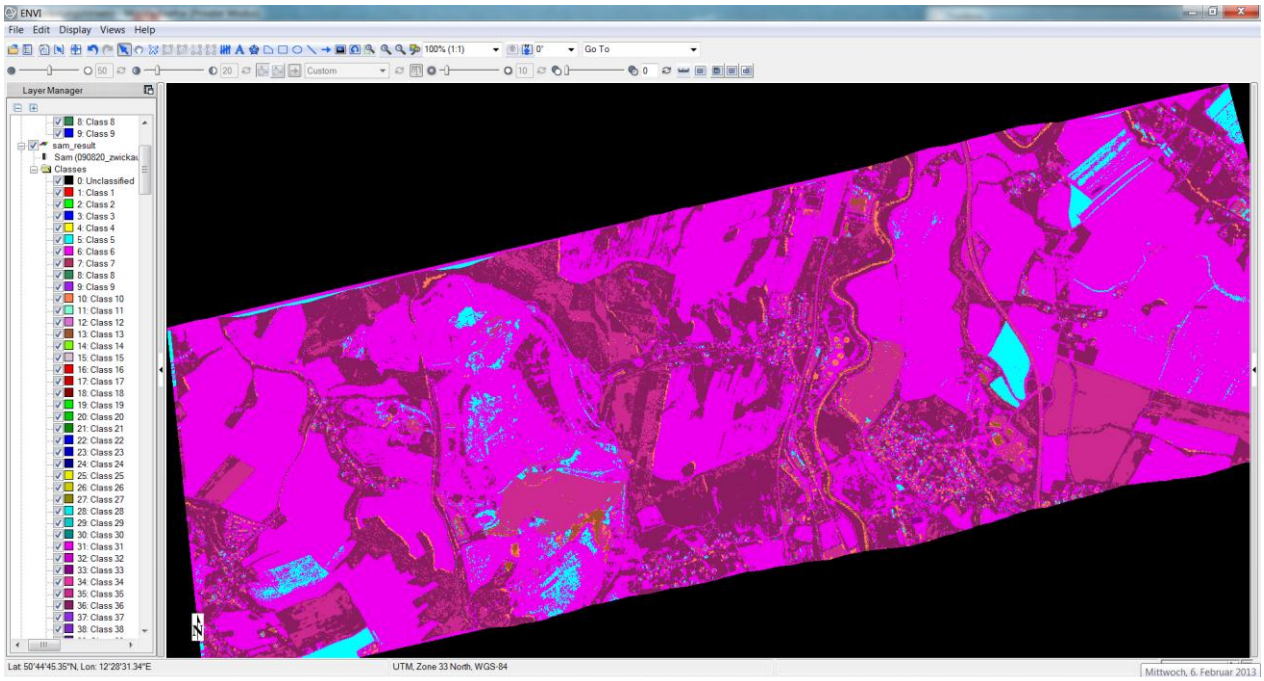


**Figure 4:** SAM classification using all acquired field spectral reflectances (reference spectra)

Therefore, we looked at spectral angles of nine classes of reference spectra (1/white: white reference; 2/yellow: reed; 3/green: bed load; 4/cyan: dark grey soil; 5/magenta: dark gravel; 6/red: red soil; 7/dark: red/grey soil; 8/dark green: gravel; 9/blue: water). Figure 5 shows the resulting SAM classification. Some classes, such as 2/yellow (reed) and 3/green (bed load) are far too overestimated, while others, such as 4/cyan (dark grey soil) and 9/blue (water) show reasonable results. The accuracy assessment will provide quantitative accuracy measures. So far, neither spectral anomalies nor significant spectral patterns could be estimated.

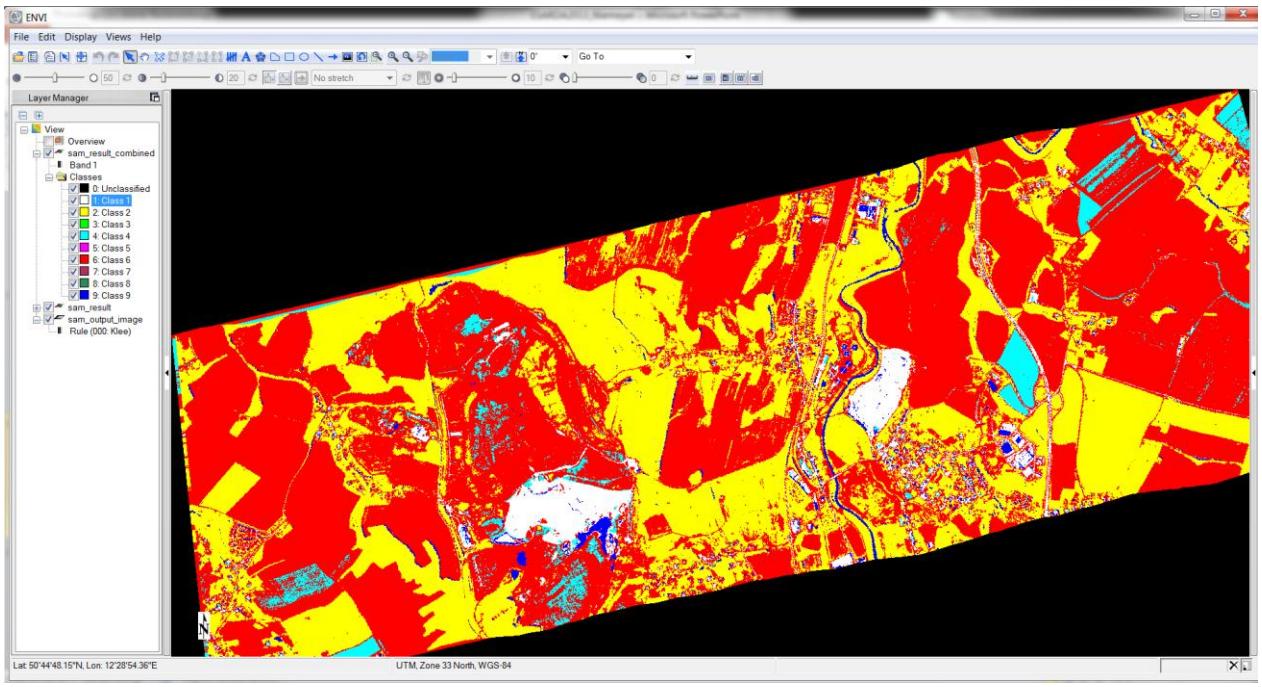


**Figure 4:** SAM classification using nine reference spectra, see text for more information.

### 3.2. Matched Filtering

Figure 5 shows the matched filtering result for one exemplary spectrum (clover). As described in Section 2.2, white pixels illustrate a perfect, black pixels the lack of match to the reference spectrum, while bright grey values explain a higher relative degree of than dark grey values.

Figure 6 displays the results for all given reference spectra. Again, a clear visualization of all (> 150) spectra becomes infeasible. We therefore focused on the 24 most prominent spectra (Figure 7), but were not able to identify spectral anomalies or significant spectral patterns anyway.

## 4. Conclusions and next steps

Neither the unsupervised processing in the first project [11] nor the supervised image processing using the reference spectra acquired in the field in the second phase, allowed for the identification of spectral anomalies or significant spectral patterns due to the former uranium processing in this region. Whether this was caused by an actual absence of contaminations in the area of interest or the application of unsuitable methods when processing the data still needs to be addressed. The third phase will therefore further integrate hyperspectral remote sensing and in-situ measurements by taking also into account the results from analysing the vegetation and soil samples using ICP-MS (Figure 8). At the end, we aim to draw conclusions on the existence or absence of directional or diffuse mining-induced environmental contaminations.

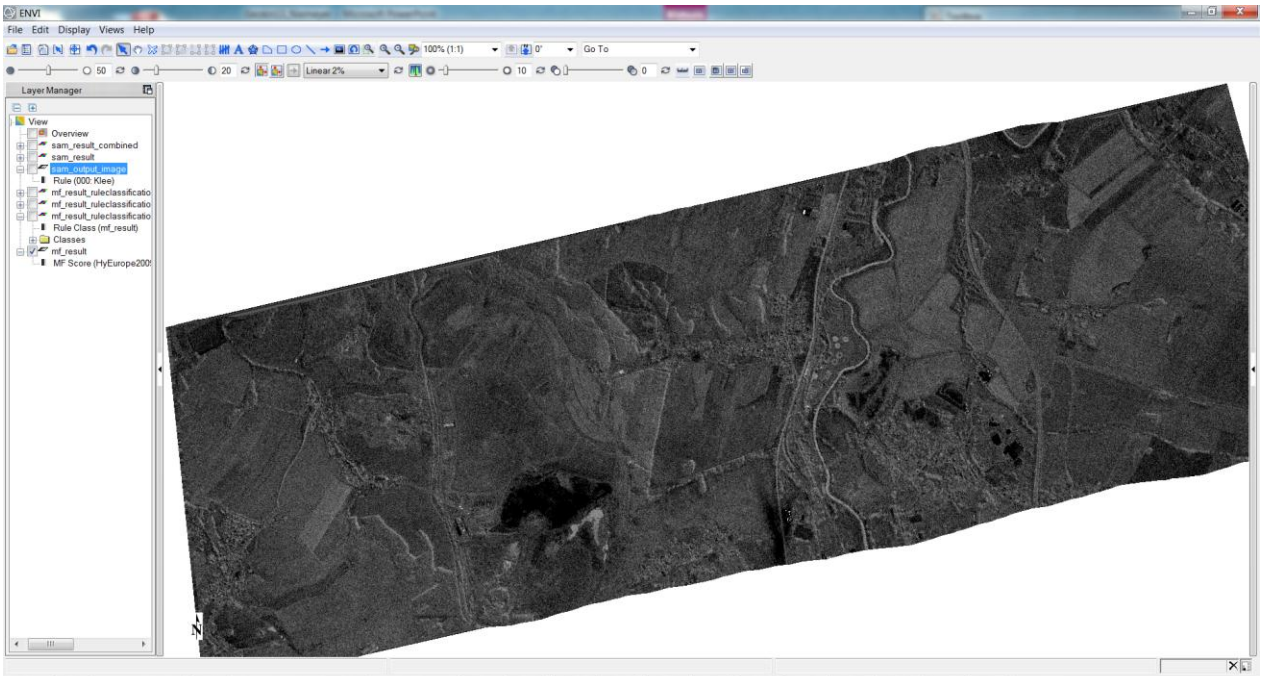


**Figure 5:** Matched filtering for the reference spectrum of clover.

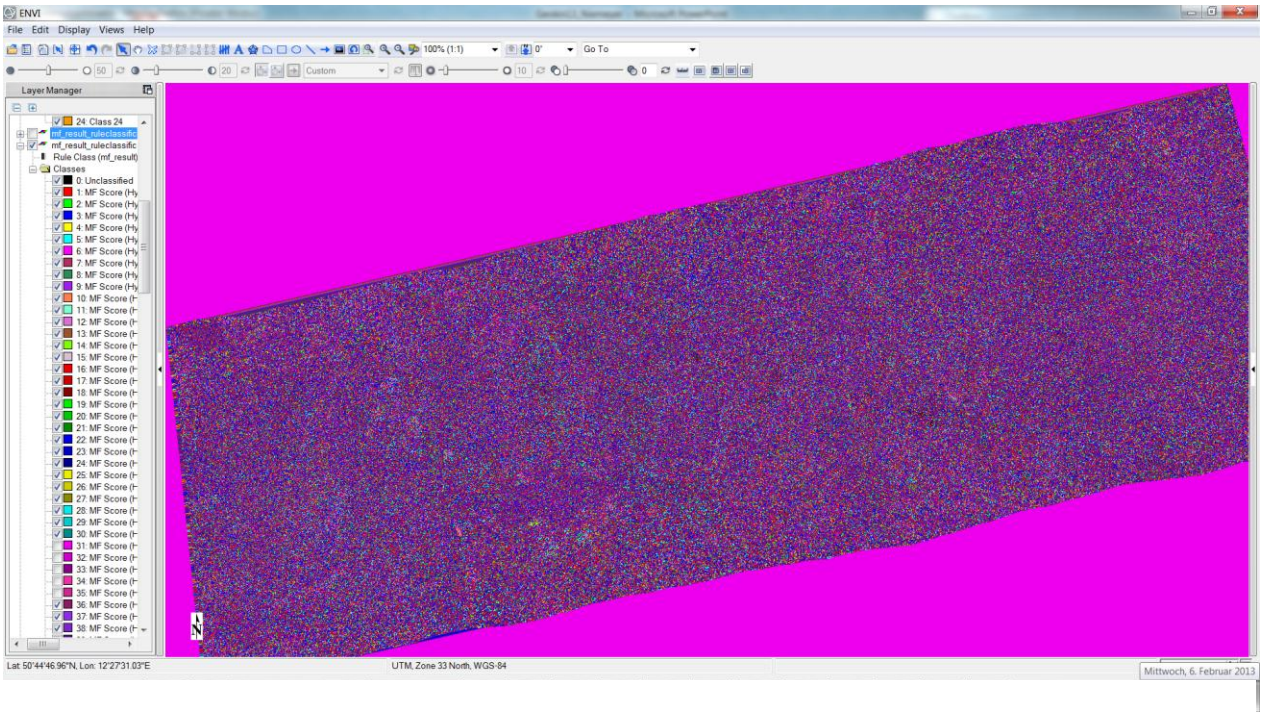


**Figure 6:** Matched filtering for all reference spectra.

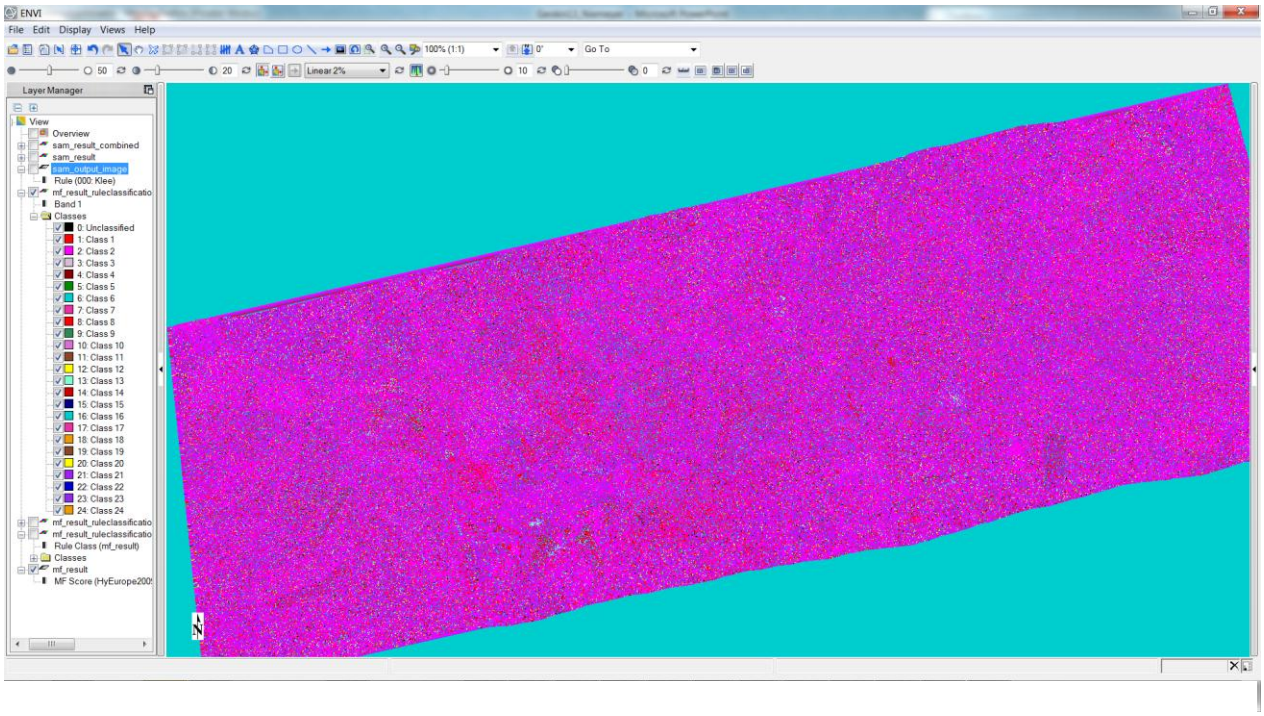


**Figure 7:** Matched filtering for 24 prominent spectra.

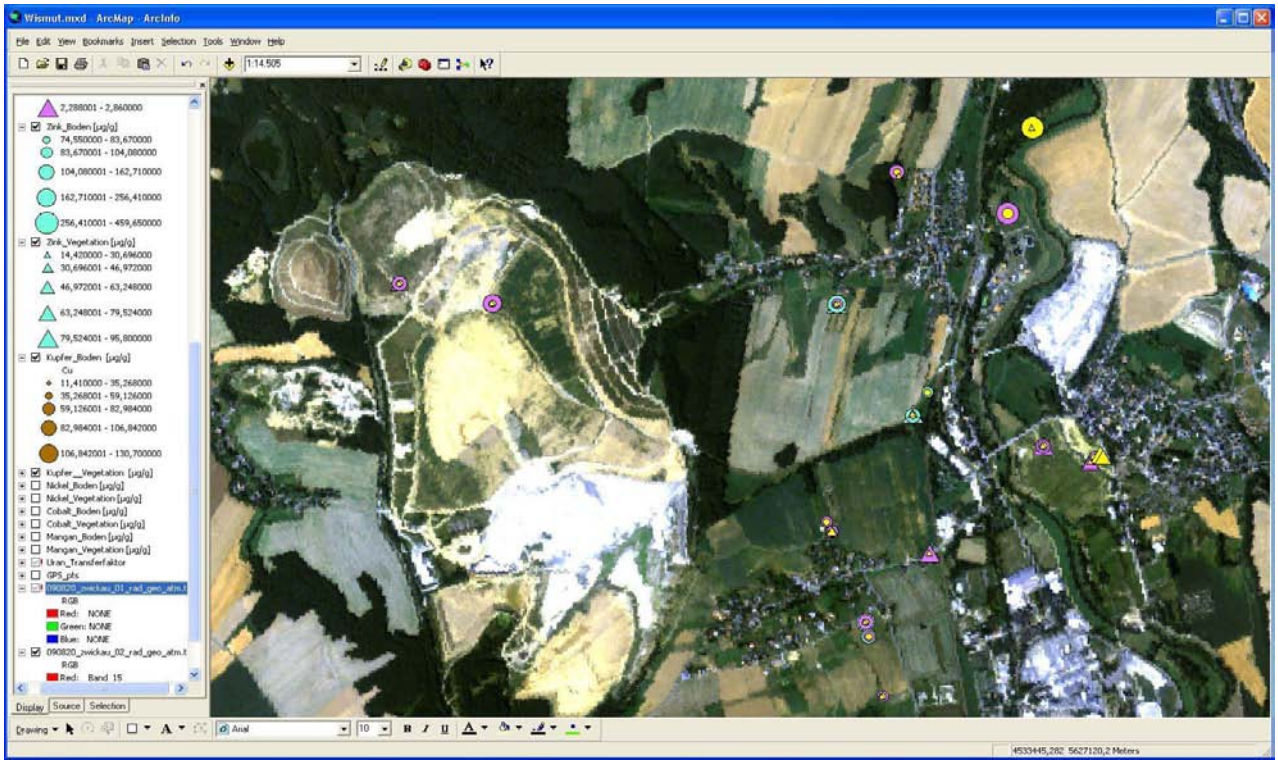


**Figure 8:** Visualization of the vegetation and soil samples analysis by ICP-MS on Arsenic, Cadmium, Cobalt, Copper, Manganese, Nickel, Uranium, and Zinc.

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# The Italian experience in implementing the Additional Protocol

**Nadia Cherubini, Alessandro Dodaro, Giorgio Giorgiantoni, Giuseppe A. Marzo, Massimo Sepielli, Francesco Troiani**

ENEA
Italian National Agency for New Technologies, Energy and
Sustainable Economic Development
Via Anguillarese 301, 00123 Roma, Italy
E-mail: nadia.cherubini@enea.it, alessandro.dodaro@enea.it,
giorgio.giorgiantoni@enea.it, giuseppe.marzo@enea.it, massimo.sepielli@enea.it,
francesco.troiani@enea.it

***Abstract:***

*The aim of the Non-Proliferation Treaty is twofold, it pursues the disarmament of the nuclear countries and, at the same time, prevents new countries from developing or obtaining nuclear weapons. In exchange these countries are granted the right of developing pacific applications of nuclear energy. The Treaty introduces new international and national Safeguard regulations and standards, and identifies the International Atomic Energy Agency (IAEA) as the organism in charge for their application and verification. The Treaty has been updated since 1973 by a number of agreements that grant IAEA complementary inspection authority to that provided in underlying safeguards agreements. In 1998 such agreements have been expanded by an Additional Protocol, ratified separately by the individual countries, which introduces Safeguard measures and controls on conventional goods and technologies, including their export to third countries, that can be also used in a military context (i.e., dual-use). The main aim of the Protocol is to enable the IAEA inspectorate to provide assurance about both declared and possible undeclared activities. Under the Protocol, the IAEA is granted expanded rights of access to information and sites.*

*Italy ratified the Additional Protocol in 2003 by means of Law No 332 on October 31 2003. Article 2 states that the Italian Ministry for the Economic Development is responsible for the civilian application of the Additional Protocol and the Ministry can delegate specific activities to other Governmental organizations. In this context the Italian Agency for New Technologies, Energy and Economic Development (ENEA) has been responsible for supporting the Ministry in several aspects of the interaction with IAEA, in providing technical and scientific support to the authority about sensitive sites, in developing technical tools, instrumentation and procedures through scientific research, in representing the country in the international context, hence playing an active role in the successful implementation of the Protocol in Italy.*

**Keywords:** non-proliferation, dual-use, sensitive trade, technology, compliance, regulations

## 1. Introduction

National and International Security is that body of provisions aimed to prevent risk for the population and the environment due to malicious actions (e.g., terroristic attacks).

Particularly after the events in 2001, security evolved for adequately responding to the new world-wide and high-technology threats. Currently, security must face different possibilities for terrorist attacks, including proliferation of weapons of mass destruction, regional conflicts, and organized crime, being one of the most significant expenses for a modern country.

Nuclear security is one of the most important aspects of security. It focuses mainly on the non-proliferation of conventional nuclear weapons and on the physical protection of the nuclear material that could be diverted for unconventional uses. The potential use of radioactive material in a terroristic attack is recent but is already perceived as a major threat despite the very different consequences following a radiological or a nuclear event.

The key element in a strategy developed in order to prevent the illicit use of nuclear material is to physically protect, possibly reduce, or even eliminate, nuclear weapons and fissile material. At the same time radioactive material must be physically protected, together with the plants and laboratories that make use of it. This is the first line of defense.

A second line of defense is to detect and track the illicit transfer of nuclear and radioactive material across international borders or inside a single country.

The third line of defense is to plan in advance an adequate response for mitigating possible illicit use of nuclear or radioactive material.

## 2. Risks deriving from the illicit trafficking of nuclear material

Illicit trafficking of nuclear material is the collection, the supply, the use, the transfer, or the non-authorized disposal of nuclear or radioactive material. By definition, it is impossible for a terrorist build a nuclear device, or a radiological dispersal device, without illicit trafficking.

Illicit trafficking of nuclear and radioactive material yields two distinct risks, 1) strongly destructive events and 2) events causing contamination or exposition. The former occurs when a nuclear weapon, technologically advanced or not, detonates. The latter when dispersion of nuclear or radiological material occurs. Consequently, the main threats are the following [1].

1. Theft of a conventional nuclear weapon. This is a military, already assembled nuclear weapon with high energy yield and high destructive effects.
2. Theft of nuclear material for assembling a Improvised Nuclear Device (IND). This is a nuclear weapon with high energy yield and significant destructive effects.
3. Theft of radioactive material for assembling a Radiological Dispersal Device (RDD). This is an explosive device used to disperse its radioactive content. Its energy yield depends on the conventional explosive adopted and its environmental impact is mainly due to the consequent radioactive contamination.
4. Theft of radioactive material for assembling a Radiological Exposure Device (RED). This is a device incorporating a radioactive source used to expose individuals transiting next to it.
5. Sabotage. This is an attack to a plant containing nuclear or radioactive material or to a vehicle transporting such materials with the aim of dispersing radionuclides into the environment. Potential targets include nuclear reactors or fuel-related plants, such as radioactive waste deposits, hospitals, industries which make use of radioactive materials.

Such risks persuaded countries to take actions for protecting the population and the environment against any event due to the criminal use of nuclear and radioactive material and, at the same time, to prevent the proliferation of nuclear weapons.

## 3. The Non-Proliferation Treaty and the Additional Protocol

In the Sixties, during the Cold War, proliferation of nuclear weapons became a matter of serious concern. In 1961 the Organization of the United Nations (UN) opened a debate about prevention measures against the proliferation of nuclear weapons. Such debate ended in 1968 with the Non-Proliferation Treaty (NPT). The NPT is currently signed by 188 countries, this means that every country conforms to it, with the exception of India, Israel, Pakistan and Cuba.

The NPT establishes a reference framework for regulating the international exchange of materials, technologies, and plants for the pacific use of nuclear energy and for implementing safeguards to prevent nuclear proliferation, the latter being the possibility for a new country to gain military nuclear

abilities. The main goal of the NPT is, in the short term, the control and, in the long term, the disarmament of those countries possessing nuclear weapons. The NPT prohibits the signatory countries to assembly, distribute, or acquire weapons, technologies, or materials potentially usable for assembly a nuclear weapon.

Nuclear technologies are allowed only for pacific purposes after the approval of the International Atomic Energy Agency (IAEA) and may be operated only under its supervision. The Safeguards Agreement signed in 1973 is the operative tool allowing the IAEA inspectorates control over nuclear materials and activities in each national territory to prevent their use for the production of nuclear weapons. This agreement has been revised several times and finally, in 1998, together with the contribution of the European Atomic Energy Community (EURATOM), it has been updated by an Additional Protocol [2], ratified separately by each country, which extends such an agreement on dual-use goods and technologies (i.e., conventional goods and technologies that could be used for military purposes) and introduces safeguard measures and controls over their export to third countries. Under the Additional Protocol, the IAEA is granted expanded rights of access to information and sensitive sites.

Overall, the goal of the Additional Protocol is to strengthen prevention by committing the signatory countries to collect detailed information on their nuclear and non-nuclear activities, even on those that do not involve nuclear materials but use components that can be adapted to activities related to the fuel cycle. Additional Protocol allows IAEA to verify if nuclear material is used differently with respect to the declarations and if undeclared material exists. In other words, the Additional Protocol extends the safeguards implemented for the nuclear material to activities related to the fuel cycle.

## 4. The implementation of the Additional Protocol in Italy

Formally, Italy ratified the Additional Protocol by promulgating Law No 332 on October 31 2003 which entered into force on April 30 2004.

Article 2 assigns to three distinct Ministries responsibility of the different aspects of the Protocol implementation:

1. Ministry of Defense is responsible for IAEA inspections in military sites, and studies, analysis and other activities related to the execution of the Additional Protocol in military sites or of military interest.
2. The Ministry of Foreign Affairs is responsible for the fulfillments of Article 11 (Designation of Agency inspectors), Article 12 (Visas), and Article 13 (Subsidiaries arrangements).
3. The Ministry of the Economic Development (MiSE) is responsible for the fulfillments of Articles 2 and 3 (Provision of information), Articles 4, 5, 6, 7, 8, 9, and 10 (Supplementary access), Article 14 (Communications systems), and Article 15 (Protection of confidential information).

Article 3 states that the MiSE assigns to the Institute for Environmental Protection and Research (ISPRA) the responsibility for the fulfillments of Article 2, letter a), and to the Italian National Agency for New Technology, Energy, and Sustainable Economic Development (ENEA) studies, analysis, and other specific activities relevant to the implementation of the Additional Protocol.

ISPRA has the fundamental role of supporting the MiSE in collecting the information required by IAEA due by the country under the Additional Protocol agreement and officially representing the Government during IAEA visits at the sensitive sites. ENEA provides support to the MiSE for several activities relative to the nuclear and conventional technology that is developed, or transit, in the country and can be potentially diverted to an illicit use.

ENEA has the expertise and the organization for the role assigned, has a history of developing nuclear and non-nuclear technology in the energy field and its official tasks include research and services in activities related to the fuel cycle such as physical and chemical characterization of nuclear materials and radioactive waste management.

Analyses of nuclear and radioactive material is carried out by the ENEA Laboratory for the Characterisation of Nuclear Materials located in the ENEA Casaccia Research Center, nearby Rome.

Additionally, this laboratory performs research on reprocessing fuel used in new-generation reactors and is the reference laboratory for characterisation of conditioned and non-conditioned radioactive waste. This laboratory has to guarantee Italy the functions of radioactive-material characterisation and process qualification by means of manipulation and radiochemical characterization of materials containing radioisotopes through gamma spectroscopy, characterization of $\alpha$- and $\beta$-emitting radionuclides, passive neutron assay and elemental analysis through fiber-optic and mass ICP. It is the National Contact Point for prevention of illicit trafficking of nuclear materials.

Radioactive material is extensively used in medical or industrial applications. Radioactive sources not adequately protected, either during their use or at the end of their life-cycle, can potentially be used for malicious purposes. Since the Eighties, ENEA coordinates an Integrated Service for the management of radioactive waste that originates from medical and industrial applications and from private and public research. The Integrated Service coordinates the collection, transport, characterization, processing, and storage of such wastes and, by means of the Legislative Decree No 52 on February 6 2007, the responsibilities of the Integrated Service have been extended to include orphan sources (i.e., An orphan source is a radioactive source which is not under regulatory control, either because it has never been under regulatory control or because it has been abandoned, lost, misplaced, stolen or otherwise transferred without proper authorization [3]). ENEA collaborates with Authorities in case of intervention.

The emergency management in the case of discovery, or suspicion of presence, of nuclear or “suspicious" material, provides a detailed list of actions to be followed against possible risks of illicit use (i.e., Security), and to ensure the protection of the population and the environment from the risks of ionizing radiation exposure (i.e., Safety).
The discovery of “suspicious" material requires the coordinated intervention of multiple agencies and organizations to carry out several actions, the main ones being the following.

1. Urgent intervention to limit the radiological risks to the population and the environment;
2. Identification and evaluation of the radioactive material;
3. Site remediation and securing the material that has to be characterized;
4. Investigation activities aimed to identify the origin of the material.

ENEA, by means of its Laboratory for the Characterisation of Nuclear Materials, carries out the qualitative and quantitative analysis of the unidentified or “suspicious" material.
The characterization can be performed by the laboratory staff with its resources and its instrumentation directly on site. If necessary, the sample can be transported to the laboratory for further measurements, destructive or non-destructive, when a full characterization cannot be achieved through mobile or portable instrumentation.

## 5. ENEA activities for the Additional Protocol

In the last ten years, ENEA made important efforts in supporting the MiSE in implementing the Additional Protocol in Italy.

At the beginning of the Additional Protocol regime ENEA realized a database of sensitive sites operating in the country and the corresponding activities that might fall under the Additional Protocol agreements. In this phase, in absence of the relative application decree, ENEA acted as a coordinator and supporting partner for those Italian subjects that were required to file the declarations in compliance of the Additional Protocol. The first declaration was sent to IAEA on October 24 2004, six month after the entry into force of the Additional Protocol in Italy. At the same time ENEA was the mediator between these subjects and EURATOM for the definitions and formalization of the new safeguard regime. Organization and support for granting complementary access to IAEA inspectors was also an ENEA task in the initial phase of the Additional Protocol implementation.

A continuous interaction with IAEA and EURATOM allowed ENEA to officially translate and distribute the localized version of the Commission’s Additional Protocol Editor (CAPE), a dedicated software used by sites representatives for preparing declarations in compliance of Articles 2.a.i (i.e., Each State shall provide the Agency with a declaration containing […] A general description of and information specifying the location of nuclear fuel cycle-related research and development activities not involving nuclear material carried out anywhere that are funded, specifically authorized or controlled by, or

carried out on behalf of, the State concerned) and 2.a.iii (i.e., Each State shall provide the Agency with a declaration containing […] A general description of each building on each site, including its use and, if not apparent from that description, its contents. The description shall include a map of the site).

On May 5 2005 an inter-ministerial application decree provided guidance for those who were required to file compliant declarations. At this point ENEA partially moved to providing support to MiSE for declarations due following Article 2.a.ix (i.e., Each State shall provide the Agency with a declaration containing […] The following information regarding specified equipment and non-nuclear material listed in Annex II) and this form of support is still one of the main ENEA tasks.

At the end of 2011 [4], all the 17 sites present in the Italian territory result compliant to the Additional Protocol regime having sent a total of 1761 report lines. They have offered 16 complementary accesses to IAEA inspectors (7% of the total European complementary accesses) who collected 20 swipes in the Italian sites so far.

At the international level, ENEA represents MiSE in several IAEA and EURATOM initiatives and working groups. In particular, ENEA has contributed to establish a line of activities in the export control of dual-use goods (EXP-WG) in the context of the European Safeguards Research and Development Association (ESARDA) [5]. ESARDA EXP-WG allows the interaction between different European public and private organizations and authorities with the aim of improving the implementation of nuclear security and keeping the competitiveness of national export regimes.

At the national level, along with its other duties in implementing the Additional Protocol, ENEA has the responsibility to support MiSE in providing training to those individuals or organizations potentially involved by the Additional Protocol regulations. Therefore ENEA organizes meetings and conference days where Authorities can illustrates regulations to stakeholders and the latters have a chance to interact with the formers for effectively improve the compliancy of their declarations.

Currently, ENEA extended its involvement in non-proliferation activities following the temporal evolution of the Additional Protocol goals. Recent efforts are coherent with the restart of the nuclear activities that have been programmed by the Italian Government, and the growing responsibility in the decommissioning activities related to the disused Italian nuclear power plants. ENEA studies new technologies and methodologies devoted to reduce the risk of diversion of the nuclear material, develops new competencies for improving safeguards, and acquires and develops new instrumentation and measurement techniques for identifying and characterize fissile material, also occurring in radioactive waste, hence playing an active role in the successful implementation of the Additional Protocol in Italy and in activities related to emergency management.

## 6. Conclusions

In the last ten years, ENEA has been responsible for supporting the Ministry of the Economic Development in several aspects of the interaction with IAEA, in providing technical and scientific support to the authority about sensitive sites, in developing technical tools, instrumentation and procedures through scientific research, in representing the country in the international context, hence playing an active role in the successful implementation of the Protocol in Italy.

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# Capture-gated portable fast neutron spectrometer

**Philip Holm, Kari Peräjärvi, Samu Ristkari, Teemu Siiskonen, Harri Toivonen**

Finnish Radiation and Nuclear Safety Authority (STUK)
Laippatie 4, 00880 Helsinki, Finland
E-mail: philip.holm@stuk.fi

***Abstract***

*Neutron detection is an important component of security and safeguards technology. A hand-held fast neutron spectrometer was developed and its response studied. The sensitive part of the detector is a borated plastic scintillator. Recoil protons caused by incident fast neutrons are registered in the scintillator. Neutron/gamma-discrimination is performed by using the constant amplitude neutron capture pulse following the scattering pulse as a gate. The detector utilizes both this direct fast neutron signal as well as an indirect high-energy gamma signal (>3.5 MeV) for neutron source detection and characterization. The background count rate above 3.5 MeV is low. Measurements with $^{252}$Cf and AmBe sources have been performed. The two neutron detection signals of the spectrometer are compared and source characterization methods are presented.*

**Keywords:** neutron spectrometer; in-field; source characterization

## 1. Introduction

Neutron detection is important in security and safeguards technology. Fast neutron spectrometry enables not only detection of the neutron source, but also source characterization. In the present paper, a hand-held fast neutron spectrometer is presented. The sensitive volume of the spectrometer is a borated plastic scintillator. Incident fast neutrons produce two pulses: a scattering pulse caused by the scattering events with hydrogen followed by a capture pulse caused by the alpha particle emitted in the neutron capture reaction by $^{10}$B. The energy of the capture pulse is 76 keVee and the pulse is used as a gate or trigger to find the scattering pulse. The height of the scattering pulse is related to the energy of the incident neutron; however, the relationship between the pulse height and the neutron energy is not trivial [1]. Unfolding the pulse height spectrum is not done in the present work. Instead, the focus is on combining the capture-gated signal with the high-energy (> 3.5 MeV) singles spectrum, which also can be used for neutron source detection [2]. Neutron sources emit high-energy photons and induce them in inelastic scattering and capture reactions. Similar measurements have been performed in Ref. [3], however, in Ref. [3], only the neutron scattering pulse height spectrum was used.

## 2. Data acquisition

The spectrometer contains a cylindrical Eljen Technology EJ254 3”x3” borated plastic scintillator (5 % natural boron by weight). The scintillator is coupled to a photomultiplier and a multichannel analyzer (MCA). The MCA is a Canberra Osprey digital MCA, which provides list mode data with a timestamp resolution of 100 ns. A minimal gain was used to obtain an energy range up to 5.2 MeV. The list mode data was analyzed with the interactive software Liisteri developed in STUK.

## 2.1 Signal processing

The time difference between the scattering pulse and the capture pulse follows an exponential distribution with measured time constant of τ = 1.6 µs. Simulations with MCNPX version 2.6f [4] proposed a time constant of τ = 1.7 µs. Due to the exponential nature of the time difference distribution, fast electronics are required for a good efficiency. Measurements showed that with its minimal signal processing parameters, the Canberra Osprey has a dead time gap of 500 ns. Another feature of the exponential time difference distribution is that the time interval preceding the capture pulse is limited by cutting the tail off.

The limit of the time trigger window was set by optimizing the minimum detectable activity (MDA). There are actually two trigger windows involved in the optimization: the time trigger window and the energy trigger window. The trigger windows are illustrated in Figure 1. Similarly to the time trigger window, the tails of the capture peak (76 keVee) are cut off and part of the neutrons are thus not registered. Combined, this leads to a trigger window efficiency that is the product of the (normalized) trigger areas:

$$\eta(k,t_m) = erf\left(\frac{k}{\sigma\sqrt{2}}\right)\left(e^{\frac{t_g}{\tau}} - e^{\frac{t_m}{\tau}}\right) \qquad (1)$$

where $k$ is half of the width of the energy trigger window, $\sigma$ is the standard deviation of the (assumingly Gaussian) capture peak, $t_g$ is the dead time gap (500 ns), $t_m$ is the cut-off limit. The parameters $k$ and $t_m$ can be optimized.

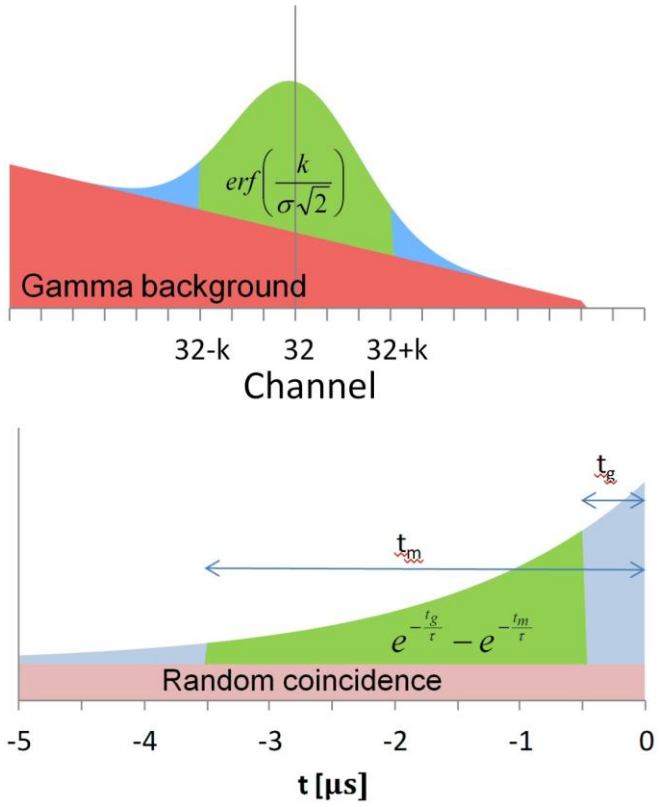


**Figure 1:** Time and energy trigger windows. See text for parameter descriptions.

The absolute detection efficiency $\varepsilon$ can then be expressed as

$$\varepsilon = \eta(k, t_m)\varepsilon_0 \quad (2)$$

Where $\varepsilon_0$ represents and ideal efficiency for which every capture-scattering-pulse pair in the scintillator is registered. ε was measured as the capture-gated count rate divided by neutron emission rate. The MDA is

$$MDA = \frac{L_D}{t_S \varepsilon} \quad (3)$$

Where $t_s$ is the measurement time and $L_D$ is the corresponding (net) limit of detection. $L_D$ depends on the user-defined type I and II false alarm rates and the background count rate, which is the random coincidence rate $N$

$$N \approx f_{cap}(k) f_{tot} \lfloor t_g - t_m \rfloor \quad (4)$$

Where $f_{tot}$ is the singles spectrum count rate and $f_{cap}$ is the energy trigger window count rate. Since the time window is on the order of microseconds, the random coincidence rate is low. For this reason, $L_D$ has to be calculated using Poisson statistics. The random coincidence count rate was lowered in the low energy region by shielding the scintillator with 1 mm of lead.

The high-energy photon signal efficiency was calculated by dividing the count rate in the high-energy region of interest (> 3.5 MeV) by the neutron emission rate.

### 2.2 Sources

$^{252}$Cf and AmBe neutron sources were used in the measurements (Table 1). A 3.8 mm thick lead shield was used with the AmBe source to attenuate the 60 keV gamma radiation. The measurements were performed at source detector distance (SDD) of 1 m. SDD was defined as the distance from the centre of the source to the surface of the detector.

**Table 1:** Neutron sources and neutron emission rates *F*.

| Source(Activity) | *F* [1/s] |
|---|---|
| Cf (20 MBq) | $2.3 \cdot 10^6$ |
| AmBe (11 GBq) | $8.0 \cdot 10^5$ |

## 3. Results

Figure 2 presents the singles $^{252}$Cf, AmBe and background spectra. The figure displays two neutron source signatures in the singles spectra: the neutron capture peak at 76 keVee (channel 32) and the significant high-energy count rate (above 3.5 MeV or channel 1378). Note also the different spectral shapes of the $^{252}$Cf and AmBe spectra due to the differences in the emitted spectra (see e. g. [5] for fission gamma spectra of $^{252}$Cf. AmBe emits 4.4 MeV photons).

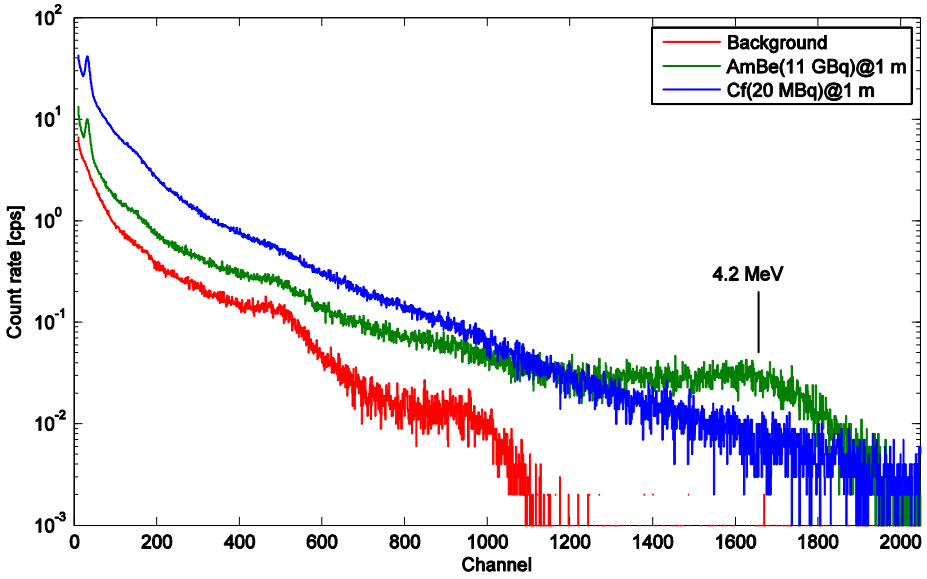


**Figure 2:** $^{252}$Cf and AmBe singles spectra measured at SDD of 1 m.

Figure 3 presents capture-gated spectra. Note that the spectra were generated with optimized trigger window widths and that these widths vary between the measurements. Table 2 summarizes the trigger parameters and detection efficiencies of the measurements.

**Table 2:** Background count rates and absolute efficiencies at SDD of 1 m. The high-energy background count rate is also included (marked width *).

| Measurement | *N* [cps] | $\eta$ | $\varepsilon_0$ | $\varepsilon_{>3.5\ MeV}$ |
|---|---|---|---|---|
| Background | 0.076 | 0.65 | | *0.10(1) cps* * |
| Cf (20 MBq) | 4.17 | 0.58 | 1.67(1) x $10^{-5}$ | 2.03(3) x $10^{-6}$ |
| AmBe (11 GBq) | 0.39 | 0.64 | 1.02(1) x $10^{-5}$ | 1.57(1) x $10^{-5}$ |

The two detection methods provide absolute detection efficiencies of the same order of magnitude. This is especially true for AmBe due to the high 4.4 MeV photon emission rate compared to the neutron emission rate.

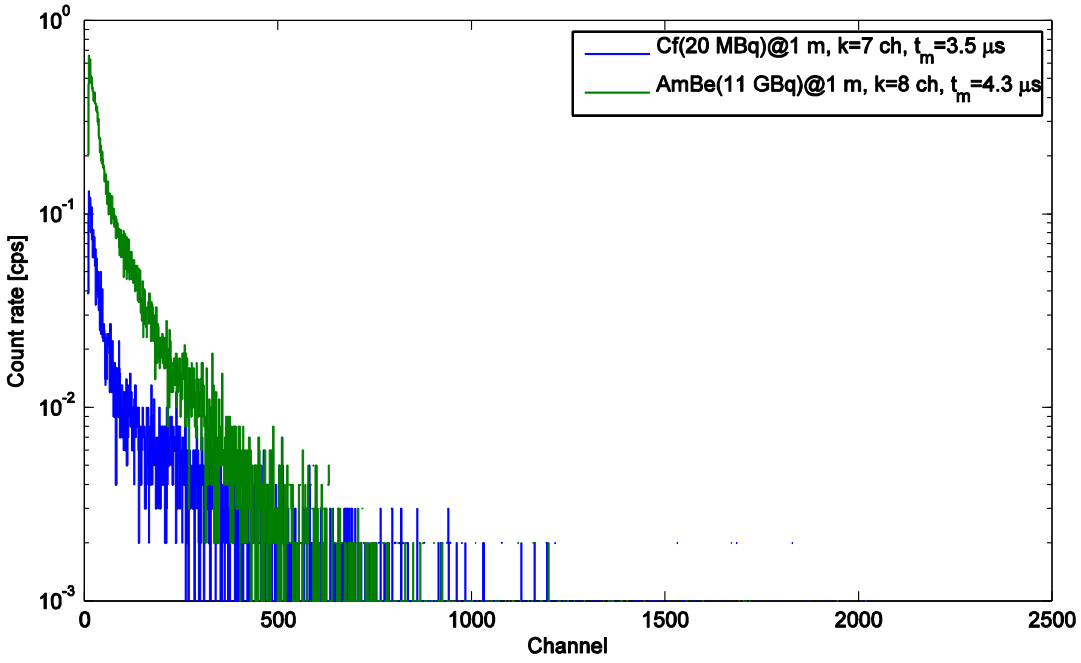


**Figure 3:** Capture-gated neutron energy spectra measured at SDD of 1 m.

One source characterization method is obvious from Figure 2. The high count rate at 4.2 MeV (the Compton edge of the 4.4 MeV peak) indicates that the source is a Be-based generator. The capture-gated spectrum provides another source characterization method. Least square error fits were performed on the capture-gated spectra to the equation $a\exp(b\,x) + c$, were $x$ is the channel number and $a$, $b$ and $c$ are fitted constants. Only data above channel 150 (about 400 keVee) was used, since the slopes below this were very similar. The fitted slope was $b$ = -0.0085 ch$^{-1}$ for the $^{252}$Cf source and $b$ = -0.0037 ch$^{-1}$ for the AmBe source. It is thus possible to discriminate the two sources by using the slope of the energy spectrum.

## 4. Summary

A versatile fast neutron spectrometer was constructed. Two neutron source detection and characterization methods were presented: the high-energy singles spectrum and the capture-gated spectrum. The methods are complementary, since both can be used for detection and characterization.

*Acknowledgements*

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# Schematic design and safeguards instrumentation of a Gen IV fuel recycling facility

**M. Åberg Lindell[1], S. Grape[2], A. Håkansson[3], S. Jacobsson Svärd[4]**

Department of Physics and Astronomy, Uppsala University, Box 516, SE-75120 Uppsala, Sweden
[1]E-mail: matilda.aberg-lindell@physics.uu.se
[2]E-mail: sophie.grape@physics.uu.se
[3]E-mail: ane.hakansson@physics.uu.se
[4]E-mail: staffan.jacobsson_svard@physics.uu.se

***Abstract:***

*The sustainability criterion for Gen IV systems, inherently presumes the availability of efficient fuel recycling capabilities. Research activities concerning advanced fuel recycling are currently pursued, and one area for such research concerns safeguards aspects of recycling facilities. Since a recycling facility may be considered as sensitive from a non-proliferation perspective, it is important to address these issues early in the design process, according to the principle of Safeguards By Design.*

*Presented in this paper is a suggested safeguards approach for a fuel recycling facility belonging to a small Gen IV lead-cooled fast reactor system that is under study in Sweden. A schematic design of a small-scale recycling facility, where actinides are separated using group actinide solvent extraction, is put forward. Measurement points are suggested based on available information on the recycling process activities and calculated material flows.*

*Based on the identified need for measurements in the facility, possible techniques and instrumentation for measurements have been identified with the purpose to provide both inspecting parties and facility operators with necessary information for their respective needs. More generally, this type of analysis may be used to support Safeguards By Design in the planning of new recycling facilities.*

**Keywords:** Gen IV; recycling; group actinide extraction; instrumentation; Safeguards By Design

## 1. Introduction

The Electra Fuel Cycle Centre (Electra FCC) is a Swedish fuel cycle concept including a 0.5 $MW_{th}$ lead-cooled reactor, Electra [1], with adjacent recycling and fuel fabrication facilities. The main goal for the Electra reactor is to accommodate education and training activities. (Pu, Zr)N is used as a reference fuel material, but also minor actinides (MA) bearing nitride fuels are foreseen.

At present, the Swedish fuel cycle does not involve recycling of nuclear fuel, and no stocks of domestic pure Pu are available. Used UOX fuel from LWR operation is stored in the Swedish interim storage for spent fuel, Clab, waiting for encapsulation and shipment to the final geological repository. However, the recycling and fuel fabrication facilities of Electra FCC are planned to *(1)* provide the reactor with fuel by separating Pu from used fuel at Clab using the Purex process, and *(2)*, in due time, recycle used nitride fuel from the Electra reactor using Ganex. In this paper, the two operation modes will from now on be referred to as *Phase I* and *Phase II*, respectively. The fuel cycle centre will, apart from providing Electra with fuel, contribute to the research in chemical group actinide extraction in Gen IV fuel cycles.

For a non-nuclear weapon state such as Sweden, the handling of pure plutonium is a new element in the fuel cycle. Hence, safeguarding the Electra FCC will present a challenge to both operator and licensing authorities. Both the facility operator and the

inspecting parties (IAEA, Euratom and the Swedish Radiation Safety Authority, SSM) must be provided with all the information they need. The safeguarding of all facilities benefit from involving Safeguards By Design [3] early in the design phase, a fact that may be even more emphasised when dealing with a facility like Electra-FCC, which will change its operation regime when moving from Phase I to Phase II. The transition must be smooth, with all the necessary equipment in place or ready to be introduced.

It is important to take the opportunity with this small facility to research not only reactor operation and fuel recycling, but also new safeguards approaches and instrumentation, and aim to reduce uncertainties in the material accountancy. In a larger scale facility, lessons learned from the demonstration facility could then be implemented in a way that is well suited for the fuel cycle.

In a previous paper [2], the TOPS methodology was used to assess the proliferation resistance of a small-scale Gen IV system including a lead-cooled fast reactor, reprocessing facility and a fuel fabrication facility. The TOPS methodology provides a guide on how to evaluate the proliferation risks of different isotope mixes and chemical forms of nuclear material. Resulting from the study was the identification of weak points and vulnerable materials in the recycling facility, where one may need to pay extra attention to the safeguards implementation. Not surprisingly, the reprocessing part of the fuel cycle turned out to offer the lowest barriers to proliferation, considering the materials included in the process. The Ganex process was shown to be, in general, more proliferation resistant than Purex, mainly due to the reduced attractiveness of materials when mixing Pu with minor actinides. Thus, Phase I is considered to pose the largest risk to diversion of nuclear material. However, Phase II is also challenging in the sense that the currently existing safeguards is not fully adapted to a Gen IV fuel cycle, with its unique material flows and properties. In this paper, Phase I is emphasised.

In this paper, a schematic design of the *recycling part* of the Electra FCC concept is provided. The schematic design of the facility includes Material Balance Areas (MBA) and Key Measurement Points (KMP), chosen from the employed recycling methods and expected material flows.

## 2. Safeguarding the Electra FCC recycling facility

### 2.1 Materials

Out of the material types that are encountered in the envisaged facility, pure plutonium is the most attractive type from a proliferator's point of view. The plutonium composition is reactor grade (RPu), but the possibility of using RPu to construct a weapon cannot be excluded [4]. By mixing the Pu with other elements, such as uranium or americium, the material needs further conditioning in order to become weapons-usable. Moreover, an increased amount of heat and radiation emanating from actinides in the mixed material may act as a barrier to theft, due to difficulties in handling. Therefore, it is advised that pure Pu streams are avoided in as large extent as possible. In Electra FCC, the plan is to blend americium into the plutonium during Phase II of its operation. (Also during Phase I, a small amount of $^{241}$Am will appear in the fuel due to the beta decay of $^{241}$Pu, having a half-life of 14 years.)

Today's timeliness goals state that the safeguards system must be able to detect the abrupt diversion of 1 significant quantity (SQ) of nuclear material during a time span of 1, 3 or 12 months, depending on the type of material diverted (unirradiated direct use, irradiated direct use and indirect use material respectively) [5]. In the case of Pu, this means that there is a minimum requirement that diversion of 8 kg Pu undergoes detection within one month. For a facility with the very small throughput of Electra FCC, 1 SQ/month is a substantial amount of material. With a throughput of merely 200 kg HM per month, out of which 2 kg being Pu, this means that four times the total throughput of Pu could be diverted each month before the timeliness goal is breached. Clearly, reaching the timeliness goal should not be a problem for the recycling facility at Electra FCC. Nonetheless, since the safeguards system for the Electra FCC may be used as a template for safeguards in larger facilities, which cannot rely on small throughputs to become proliferation resistant, the suggested safeguards instrumentation should be able to give adequate results.

### 2.2 Key Measurement Points and Material Balance Areas

The IAEA utilises Key Measurement Points and Material Balance Areas for accountancy purposes. A KMP is defined as "a location

where nuclear material appears in such a form that it may be measured to determine material flow or inventory" [5]. One may partition the KMPs into two groups; inventory KMPs and transfer KMPs.

One may select to define MBAs of a facility in different ways. Which MBA definitions are selected depends largely on practical issues such as the possibility to obtain accurate material balances, and thus on the amounts and forms of material present, and also on the possibility to determine the quantities of material transfer into or out of the MBA. Material balance areas are areas in or outside of a facility such that the quantity of nuclear material in each transfer into or out of each MBA can be determined [5]. An operator can use an extended amount of MBAs for their internal material accountancy, but officially report their accountancy based on a reduced amount of MBAs to the IAEA, depending on the Agency's requests.

In this paper, the Electra FCC recycling facility has been divided into five MBAs, such that detailed records of the material accounts can be kept in each part of the recycling process. The MBA distribution presented also serves to resemble the approach for a large scale reprocessing plant which, due to its complexity, often requires several MBAs [6]. It would however be likely that the IAEA perceives Electra FCC as one single MBA, due to its small size and throughput. This does not, however, change the implementation of safeguards instrumentation within the facility.

### 2.3 The needs for safeguards measurements

The overarching safeguards goal is to minimise the material unaccounted for (MUF) in the facility, by carefully measuring properties of the materials coming into the facility as spent fuel, and going out of the facility in different streams as end products or waste. Also material going in and out of each MBA should be measured. The measurements are performed at inventory and transfer KMPs. Transfers within MBAs are not relevant, but transformations of material are (different chemical forms, isotopic compositions etc.). In order for Electra FCC to function as a template for large-scale facilities, it may be wise to consider redundant KMPs. Thus, cross-checks between KMPs can be implemented, and predictions of measurement results at can be made.

In addition to the key measurement points, it is desirable to follow the nuclear materials throughout the facility such that one may, in the event of a diversion, localise possible diversion routes. Furthermore, if the facility operation has to be shut down due to e.g. equipment failure, and the dissolved material has to be flushed back to a previous step in the separation process, one must not lose the continuity of knowledge. In a larger scale facility, a solution monitoring system (logging tank levels and fluid densities, and transfers between vessels using neutron sensors) with near real time accounting may even be required in order to fulfil the timeliness goals [7]. Since Electra FCC can be seen as a demonstration of the safeguards system for a larger plant, adding solution monitoring capabilities is advisable.

## 3. Suggested measurements in the Electra FCC facility

As mentioned in section 1, the acquired design presupposes two modes of operation to be run at the fuel cycle centre [8]. In Phase I of Electra FCC's operation, oxide fuels will be reprocessed using the Purex process, in order to obtain fresh plutonium nitride fuel for the first cores of the Electra reactor. Once the reactor is up and running, used nitride fuels will be recycled at the FCC (Phase II), see fig. 1. A fuel fabrication facility will be connected to the recycling part of Electra FCC. In Phase II, the Ganex process will be used to create new minor actinides-bearing nitride fuels for the Electra reactor. A sol-gel technique, where the Pu solution is converted to microspheres instead of powder, will be used for the fuel fabrication.

During Phase I, Electra FCC will be similar to existing reprocessing plants using Purex. Therefore, studies of the implemented safeguards approaches at currently existing reprocessing plants have been useful. Extra attention has been paid to Japan, being the only non-nuclear weapon state pursuing reprocessing of used nuclear fuel. However, comparison with e.g. the Rokkasho Reprocessing Plant (RRP) is not relevant in all aspects, partly due to the large differences in throughput (RRP: 800 MTHM/y).

The types and quantities of nuclear material present in a facility largely define the needs for nuclear safeguards and affect the operation scheme of the facility. Some assumptions have been made in order to estimate material flows in Electra FCC. Below are descriptions of the

operations of Phase I and Phase II respectively, together with declarations of the materials that flow through the different parts of the recycling facility, and suggested measurements.

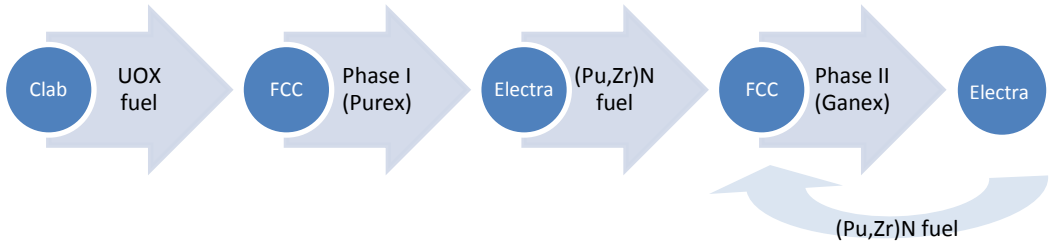


**Figure 1:** Phase I and Phase II in the Electra FCC recycling process involve different fuel types as well as different reprocessing techniques.

## 3.1. Phase I

The Electra core encompasses 397 fuel pins, each containing 0.17 kg Pu, and needs about 68 kg of Pu initially [9]. This amount of Pu should be acquired from used Swedish UOX fuel currently residing in Clab. It is assumed that the fuel needed has an average TRU vector of: 3.5 / 51.9 / 23.8 / 11.7 / 7.9 / 1.2 ($^{238\text{-}242}$Pu/$^{241}$Am). The full core should be inserted in the reactor in, at best, the year 2023. The FCC facility is here assumed to be in operation in 2019, which gives 4 years to produce the necessary amount of Pu. Assuming that the recycling facility will be able to run at its full capacity this whole time, and that the fuel at Clab contains 1 % Pu, 1700 kg of spent fuel will have to be recycled yearly during four years, producing an output of 17 kg Pu/year. Thus, a total of 6.8 tons of spent nuclear fuel from Clab will be needed to fuel Electra.

Batch-wise operation at Electra FCC is the most practical option, with one batch (= one fuel assembly) being dissolved each month. Thus, no more than one fuel assembly should ever have to be present in the head-end facility storage. This is advantageous from a safeguards perspective due to the small amount of nuclear material present, and due to the fact that mixing of several fuel assemblies in one batch is avoided. Following material streams from the dissolved spent fuel is straight-forward when only one assembly is dissolved at a time. Some degree of mixing is inevitable unless system flush outs are performed in between every batch, but the holdup volumes residing in e.g. pipes will be calculated.

One fuel assembly (approximately 200 kg HM) is transferred to Electra FCC every month. The assumed composition of the fuel is 5 % fission products (FP) and 95 % actinides, whereof 1 % is Pu (isotopic composition mentioned above), 0.1 % Am, 0.01 % Cm, and the remaining part U.

Figure 2 depicts Phase I of Electra FCC's operation. The arrows represent the flow paths of different constituents of the nuclear fuel. The five rectangular areas represent material balance areas. The operations and material inventories of the MBAs are described below. For the head-end and separation MBAs, key measurement points are suggested, whereas the sol-gel, waste and U storage MBAs are investigated in less detail.

***Head-end MBA***

The head-end MBA is depicted in figure 3. Here, spent fuel is received and immediately put into a storage pond for shielding. The used fuel is then sheared, i.e. chopped into pieces having a length of a few centimetres. The fuel pieces are dissolved in hot nitric acid ($HNO_3$). At the end of the head-end area, the dissolved spent fuel is transferred to an input accountancy tank.

Inventory:
- Spent UOX fuel from Clab
- Sheared fuel
- Fuel dissolved in nitric acid
- Hulls and end pieces. Hulls separated from the dissolution stage will contain traces of actinides.

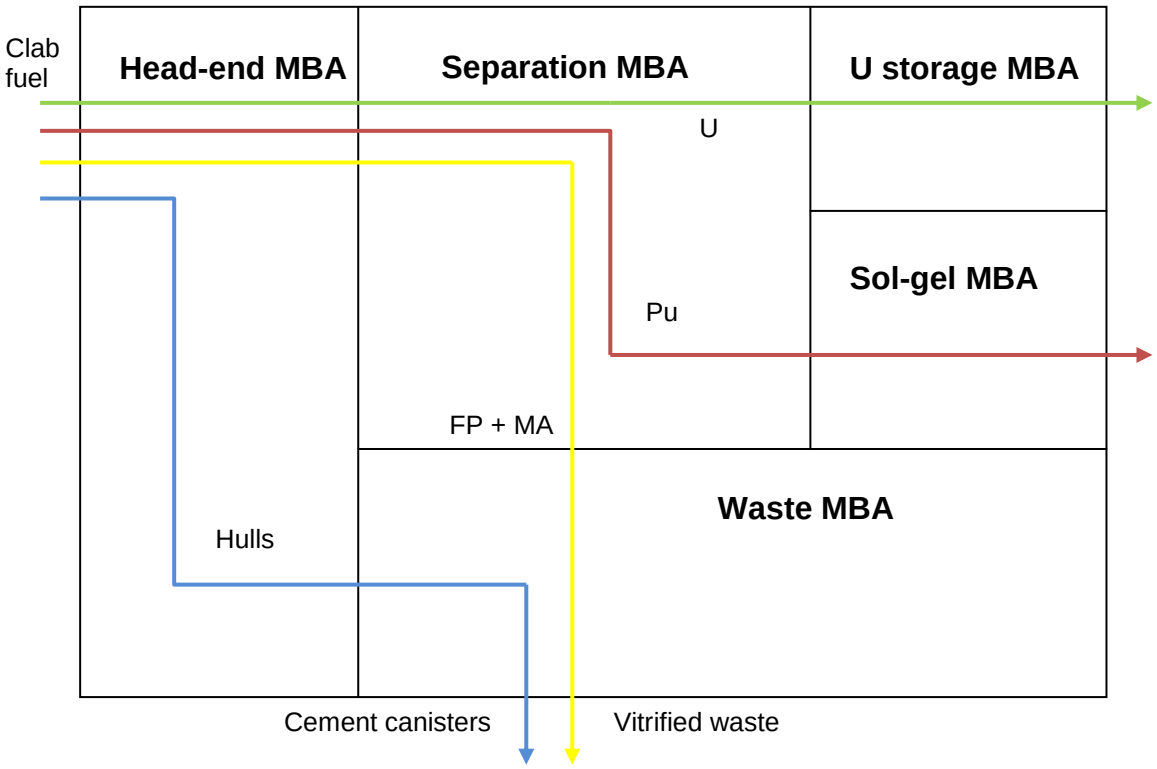


**Figure 2:** Schematic design of the recycling in Electra FCC, Phase I.

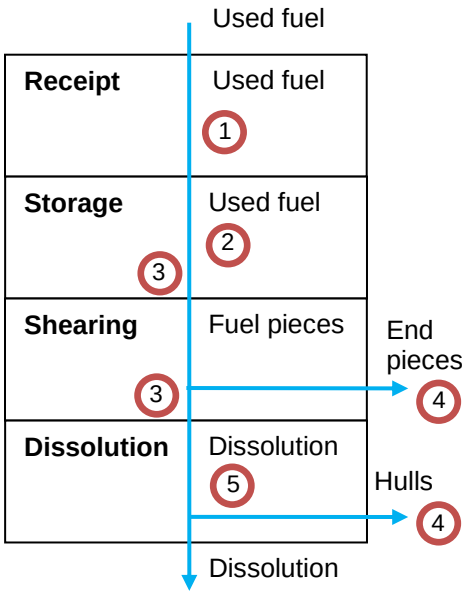


**Figure 3:** Suggested key measurement points in the head-end MBA of the Electra FCC recycling facility, Phase I.

The head-end area is a very important MBA, since possible errors in the measurements here (or a verification of inaccurate spent fuel data) will cause the whole facility's accountancy records to suffer from erroneous data, and the sum of the outgoing materials will not match the nuclear material masses in the incoming fuel. Once the fuel has been dissolved, turning into bulk material instead of items, it becomes more challenging to safeguard. The last opportunity to carefully measure the fuel as an item should therefore not be missed.

The numbered KMPs below are marked in fig. 3.

1. At the receipt area of spent fuel, item counts and ID checks of the incoming assemblies are necessary.
2. Cameras can be used to keep continuity of knowledge in the storage space for used fuel. Neutron and gamma based NDA measurements of the fuel should be carried out in order to obtain masses of fissile material in the fuel [10]. Here, there is an opportunity to improve currently employed methods with respect to measurement accuracies (an estimation of the current level of accuracy for Pu and U is 10-20 % [11]). Using a small facility such as Electra FCC for testing novel safeguards equipment is recommended. Also advanced

measurement techniques such as gamma tomography could be employed to detect partial defects at this stage.

3. Continuity of knowledge must be kept in the storage space for used fuel and during transfer through the shearing and dissolution processes. Transfers could be verified with cameras and radiation monitors [7].
4. The amount of residual fuel content in hulls and end pieces must be determined, in order to assure that the dissolution has been successfully performed, and that no nuclear material is being diverted through the hulls waste route. This can be achieved using gamma spectroscopy and neutron counting techniques [10].
5. Measurements of the dissolution in the input accountancy tank are needed to verify that all the incoming fuel has been dissolved. DA samples should be taken from the dissolution for high precision measurements of Pu and U, but also Am and Np (for process quality control as well as for voluntary reporting to the IAEA). K-edge and X-ray fluorescence spectrometry may be used for the analyses [10]. The measured quantities would help to evaluate the accuracy of the NDA measurements at KMP 1.

***Separation MBA***

The separation MBA is where uranium and plutonium are separated from each other and from the fission products, see fig. 4. During Phase I, this will be done using the well-established aqueous reprocessing technique Purex [12]. Extraction columns, small enough to prevent criticality, will be used for the separation. It is assumed that the separation techniques are 99.9 % efficient, meaning that 0.1 % of the plutonium will end up in the waste streams. Uranium undergoes conversion to oxide powder. Fission products and minor actinides left in the dissolution are headed for the waste treatment MBA, whereas the uranium and plutonium are taken care of in separate MBAs. The pure Pu stream arising in the separation process is the main concern in the MBA.

Inventory:
- Feed
- Dissolved uranium
- Dissolved plutonium
- Dissolved waste products (FP and MA with traces of U and Pu)
- Uranium oxide powder

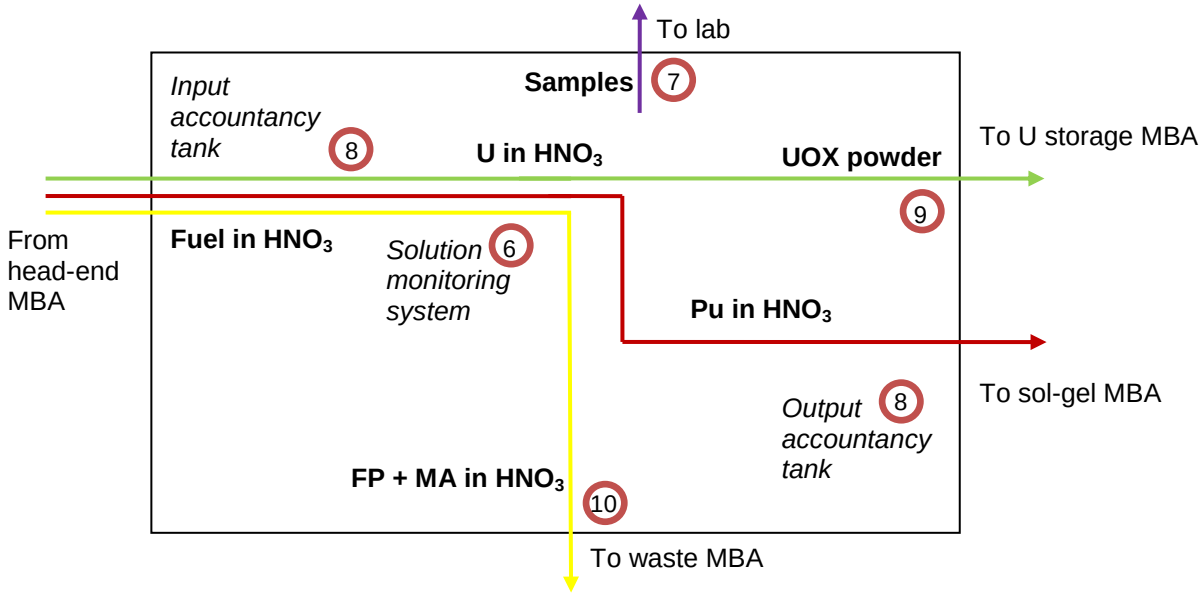


**Figure 4:** Suggested key measurement points in the separation MBA of the Electra FCC recycling facility, Phase I.

The separation MBA is shown in fig. 4. The numbered KMPs below are marked in the figure.

6. A solution monitoring system measuring tank levels, densities and neutron counts is a good means for assuring that the process flows and facility operations are as declared [7]. All transfers between vessels should be monitored by the system.
7. Sample taking is a necessary part of the safeguards system at the facility, supplemented by NDA measurements of the solution monitoring system. An on-site laboratory would most likely not be available for a low throughput facility such as Electra FCC due to the large resources needed to run it. In a large facility, though, an on-site lab may be required.
8. Contents of the input and output accountancy tanks should be measured and compared to each other to verify that no material has been added or gone missing in the separation stage.
9. The enrichment of uranium converted to oxide may be verified with DA sampling. C/S measures should be applied to the oxide powder.
10. Liquid waste volumes should be measured, and DA samples taken to confirm the U:Pu:$^{244}$Cm ratio [7].

***U storage MBA***

Depleted uranium product from the spent fuel will most likely be stored at the facility for a short period of time before being shipped to another storage facility for either subsequent MOX fuel fabrication or final disposal. It is assumed that the U will be stored in the form of UOX powder, which is commonly produced in other facilities today.

Inventory:
- Uranium oxide powder in containers

The stored uranium is not as desirable to a potential diverter as the Pu, but it must be accounted for. A combination of weighing and destructive assay is suggested, to measure the U and $^{235}$U content [13]. Possible plutonium contents in the U storage containers could be measured using neutron and gamma based NDA methods [10].

***Sol-gel MBA***

The solution containing the separated plutonium will be manufactured into microspheres in a sol-gel microsphere pelletisation (SGMP) process [14]. Large amounts of these small spheres show a fluid-like behaviour, which enables transport in pipes. The spheres do not form dust, and therefore cause less contamination in the facility than the regular powder product. The microspheres are transferred to the fuel fabrication facility, to be pressed into pellets and sintered. The fuel fabrication facility is, however, not considered in this work.

Inventory:
- Dissolved plutonium
- Plutonium microspheres

The monitoring of nuclear materials in the form of microspheres is new to Sweden, but is not foreseen to cause problems. The same radiation measurement techniques as those used to monitor the dissolved fuel should apply here as well, considering the fluid-like behaviour of the microspheres.

***Waste MBA***

Several different waste streams are present in a recycling facility; fission products and minor actinides in a nitric acid solution, cladding materials, and waste from the liquids used in the recycling process. These will all be treated as waste in the same MBA. Hulls and end pieces are packed and put into canisters. Liquid waste is dried/solidified and placed in cement in canisters. The acid containing FP and MA enter an accountancy tank before being vitrified and put into containers.

Inventory:
- Hulls and end pieces
- Dissolved FP and MA
- Vitrified waste
- Waste canisters

The main concern in the waste area is to confirm that no plutonium is hidden in waste such as hulls, end pieces and liquid waste. Neutron measurements applying e.g. the differential die-away technique should be applied to ensure that no more than small traces of nuclear material will be present in the waste streams [10]. Neutrons emanating from $^{244}$Cm in the vitrified waste, together with the U:Pu:$^{244}$Cm ratio in the dissolved fuel, can be used to determine the Pu content in the waste [7].

### 3.2. Phase II

It is planned that Electra will be able to run for 9500 full power days on the initial fuel obtained from the reprocessing in Phase I, without refuelling. However, this scenario would not allow for testing MA-bearing fuels in the reactor, nor recycling of the used nitride fuel, which would be a purpose for this type of demonstration facility. Therefore, in this paper, the option of replacing fuel assemblies after a shorter period of time is investigated. It is assumed that 10 % of the pins in Electra are replaced each year. If so, approximately 7 kg of plutonium will pass through the recycling facility each year. Minor actinides and uranium isotopes are produced during the reactor operation, whereas the Pu inventory decreases.

Americium will be introduced in the new nitride fuels to avoid having a pure Pu stream. Am and Pu are extracted as a group at the same point in the recycling process, i.e. they are never separated. It is desirable to put as much Am in the fuel as possible, for both proliferation resistance and waste management reasons. No more than a few percent can be blended in, though, due to deterioration of safety parameters in the reactor [15].

Curium is not wanted in the Electra fuel, since heat and radiation would cause problems in the fabrication process. There are different suggested options for the storage of the extracted Cm, including vitrification and placement in a final repository, and a dedicated Cm storage where the Cm can stay for a couple of hundred years until it decays to Pu [16]. The assumption in this work is that no matter which one of the two options is chosen, the Cm stream will leave the Electra FCC through the waste MBA after separation from the other actinides.

Figure 5 shows a schematic of the Electra FCC during Phase II of its operation, and the types of nuclear material in the inventory are listed below.

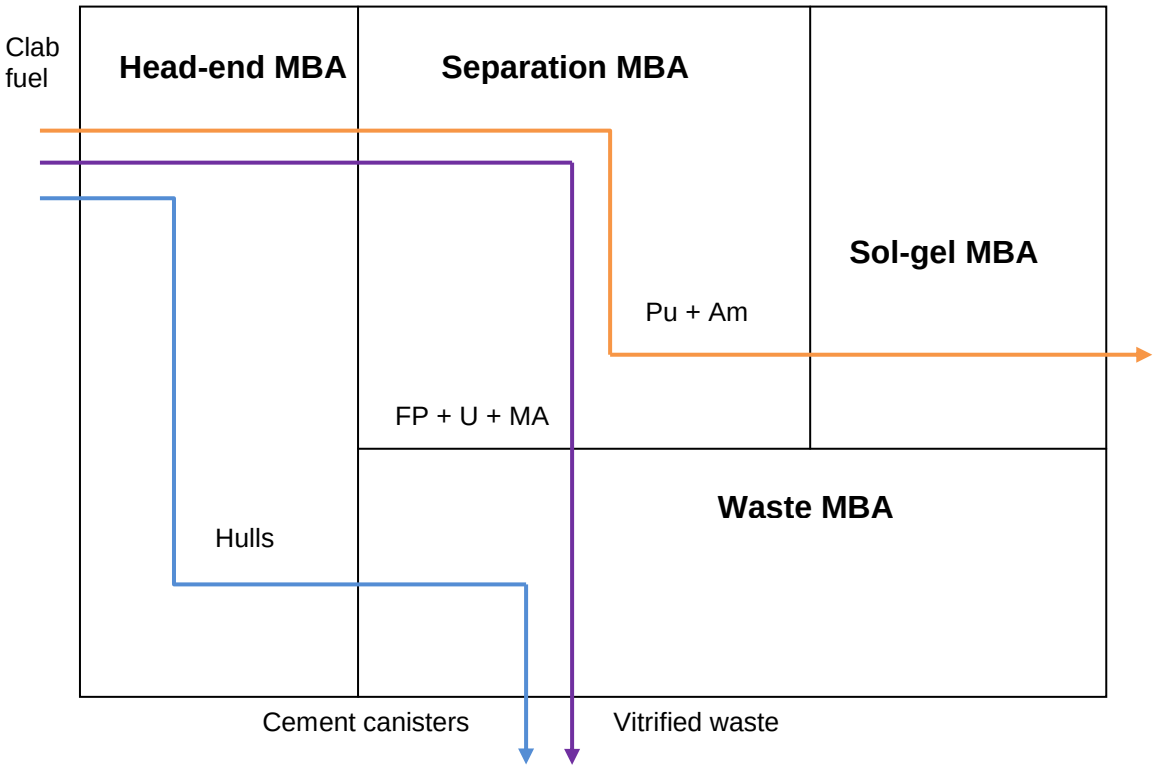


**Figure 5:** Schematic design of the recycling in Electra FCC, Phase II.

The material streams in Phase II of Electra FCC's operation differ from Phase I in a few ways. Most importantly, plutonium nitride fuel will be subject to the separation process, instead of uranium oxide. The dimensions of the Electra fuel pins differ from the UOX fuel pins from Clab, and the head-end shearing and dissolution equipment must be able to adapt to the different kinds of fuel. In the separation stage, the group actinide extraction technique

Ganex will be used, such that the plutonium is always mixed with other actinides [17]. The dilution of the plutonium reduces its attractiveness and makes the material inherently more proliferation resistant [2]. Uranium will not be separated, which eliminates the need for a uranium storage MBA.

**Head-end MBA**
- Spent plutonium nitride fuel from Electra
- Sheared fuel
- Fuel dissolved in nitric acid
- Hulls and end pieces. Hulls separated from the dissolution stage will contain traces of actinides.

**Separation MBA**
- Feed
- Dissolved Pu + Am
- Dissolved Cm
- Dissolved waste products (FP with traces of actinides)

**Sol-gel MBA**
- Dissolved Pu + Am
- Pu + Am microspheres

**Waste MBA**
- Hulls and end pieces
- Dissolved FP
- Vitrified waste
- Waste canisters

The safeguards implementation required for the two phases of Electra FCC's operation will be the same for many of the identified key measurement points. Key measurement points in the Phase II configuration of the recycling facility are expected to remain in essentially the same facility locations as in Phase I, since transfers of materials follow roughly the same routes. However, since the types of materials to measure will differ, the instrumentation for the KMPs should be revised to suit the materials in Phase II. As an example, measurement and verification of the received spent fuel might be different due to the properties of oxide vs. nitride fuels, and nitride fuel measurements techniques should be explored in detail before being implemented in the facility. Measurements adapted to the properties of nitride fuel may then be evaluated in the small facility.

## 4. Conclusions and discussion

Pure Pu streams pose, from a safeguards perspective, a challenge to the fuel cycle system. The small amount of Pu needed for use in the Electra reactor, due to the small size of the core, does somewhat mitigate the difficulties that arise with the undesirable material properties. Nevertheless the measurements performed in Electra FCC should be as accurate as possible, in order to examine the best achievable accuracies in measurements of various material types. Even though the IAEA's timeliness goals for the facility may be overreached, a demonstration of accurate measurement techniques intended for a full-scale facility should be provided. The facility operators may also benefit from improved measurements, for their solvent extraction research purposes.

The choices for instrumentation at the different measurement points should reflect the aim to pursue as accurate measurements as possible. Today's instrumentation standards will be sufficient to make Electra FCC fulfil the IAEA's timeliness goals, and existing techniques such as gamma ray spectrometry, neutron coincidence counting and K-edge densitometry can readily be used in the facility. However, in a larger recycling facility which cannot rely on a small throughput to make it proliferation resistant, possible development of the instrumentation might be crucial. It is advised that e.g. the Next Generation Safeguards Initiative (NGSI) techniques are investigated further for use in a recycling facility [18].

A small scale demonstration facility such as Electra FCC will benefit not only research on Gen IV recycling techniques, but may also act as an aid in developing a safeguards approach for future large scale facilities, by adding novel measurement equipment and evaluating the system's performance in detecting nuclear material. The low-throughput facility provides a unique environment for testing performances of new and existing safeguards equipment and data acquisition. This way, Safeguards By Design can be more easily implemented in full-size recycling facilities.

Virtual reality based safeguards design tools may be used to model new nuclear facilities in 3-D, implement safeguards equipment in the facility model and test different safeguards setups [19]. It is advised for stakeholders involved in the early design stages to take such computerised design utilities into

consideration, since they may both decrease the cost and ease the implementation of SBD in new facilities.

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# Lessons learned in implementing DCVD partial defects training

**D.A. Parcey[1], R. Kosierb[2], E. Sundkvist[3],**

[1]Channel Systems Inc., P.O. 188, Pinawa, Manitoba, R0E 1L0, Canada
[2]Canadian Nuclear Safety Commission, Ottawa, Ontario K1P 5S9, Canada
[3]Hacko-Teknik, Stockholm, Sweden

***Abstract:***

*In 2012, the Digital Cerenkov Viewing Device (DCVD) was approved for partial defects detection use by the International Atomic Energy Agency. This necessitated the development of a training program that would prepare an inspector to use the DCVD and the partial defects detection method. The development of the training course was based on the successful Spent Fuel Verification training program that has been used for many years to train the IAEA inspectors. The course is comprised of a set of lectures on the theory of partial defects detection and a practical field component at a spent fuel storage pond where the inspectors could be given hands on training and experience. The partial defects training course has now been successfully presented three times, each in a different facility under different conditions. This paper will discuss the overall layout of the training course the background work required to set up such a course and some of the lessons learned in preparing and presenting the course at different facilities.*

**Keywords: DCVD, Partial Defects, Training**

## 1 Introduction

The Digital Cerenkov Viewing Device was approved by the IAEA for the detection of partial defects early in 2012. A Partial Defect Detection (PDD) training course for IAEA inspectors was developed and the first session was given to nine IAEA staff in March 2012. There have been two additional PDD courses presented subsequent to the March 2012 course. This paper will discuss the format of the courses presented and provide observations and recommendations for others producing similar courses.

## 2 The Partial Defects Detection Training Course

The PDD course was initially based on the successful Spent Fuel Verification (SFV) course that has been taught to IAEA inspectors and others since 1988. The SFV course provides the trainees with an understanding of the process of identifying gross defects (spent fuel versus non-spent-fuel) using the Improved Cerenkov Viewing Device (ICVD), the Digital Cerenkov Viewing Device (DCVD), the Spent Fuel Attribute Tester (SFAT) and Fork Detector (FDet). The SFV course starts with three days of lectures and is followed by a more than a week at spent fuel storage ponds doing exercises designed to teach verification methods in a step by step manner. The week is concluded with a simulated inspection and a course quiz.

The PDD course is more focussed than the SFV course but it is held over four days so it is quite intense.

## 2.1 Training Objectives

The first important step in creating a training course is to determine the objectives of the course. The main goal of the course is to ensure that the trainees have sufficient training and practice to identify partial defects with the DCVD. In the case of the PDD course, there are four main objectives that must be met for a training course to be successful. These objectives are:

1. The trainees should understand how Cerenkov light is generated and how the measurement of Cerenkov light is used to detect the absence of fuel rods or the presence of substituted fuel rods.
2. The trainees must understand how the DCVD operates, how to set it up, and how to use it in different facilities and environments.
3. The trainees should be able to evaluate the environmental factors that have an impact on the measurements.
4. There needs to be an understanding of how to obtain a quality measurement and how to interpret the information and understand the measurement results.

The development of the training course must be based around a clearly defined and achievable set of objectives. When setting out the objectives for the course, decide not only how they will be taught but how they can be measured. This will allow the determination of the effectiveness of the course.

## 2.2 Planning the course

### 2.2.1 Course Schedule

Preparing the schedule for the PDD is often the most difficult aspect in course preparation. The schedule must be based on the workday of the facility to avoid overtime or conflicts. If the bridge operator takes lunch from 11:00 to 12:00 the schedule must reflect it. If the facility personnel finish at 15:30 then the instructors and trainees must be out by 15:30 or earlier.

The schedule should include the starting time, the sessions/exercises planned for the day and the wrap-up session. Don't forget to factor in time for the facility staff lunch. Also allow time for entering and exiting the facility each day.

Preparing a schedule is complex because a number of activities can be in progress at the same time. The time in the facility is very valuable and the schedule should make use of every minute available. This is the opportunity for the trainees to get practical experience in a rea-world environment so the schedule should reduce the amount of waiting time. All of the effort put into creating a workable schedule will prove valuable as the course will run more smoothly.

**Figure 1: If the schedule is carefully thought out, multiple exercises can proceed simultaneously**

No matter how well thought out the schedule is there will always be disruptions, delays and changes as the course proceeds. It is impossible to anticipate these but being aware that changes will occur can make a course much less stressful for the instructors.

### 2.2.2 Workbook

A course workbook for each course is prepared containing the course schedule, the slides for the lectures, a description and worksheet and pond map for each exercise. Each trainee is provided with a workbook at the beginning of the course and each instructor is given a workbook with the facility pond maps included. The trainees have all the reference material they need for the course in one place.

### 2.2.3 Simulation of partial defects

In reality there are very few known partial defects available for measurement in spent fuel. This means that in most cases a partial defect will have to be simulated.

It is not possible to make an assembly look like it has a lot of fuel pins missing but the DCVD can detect single missing fuel rods so an example of a small defect can usually be found in a facility with lots of fuel assemblies. Generally the facility has records of assemblies with missing fuel rods and can provide information on the location of the assembly.

Substitution scenarios have to be simulated. Since the DCVD uses the burnup and cooling time declaration provided by the facility, a substitution scenario can usually be created by modifying the facility's declaration of burnup and cooling time so it will no longer match the measurements taken by the device.

Having a group of trainees try to find a false declaration in a group of fuel assemblies is an interesting challenge for both the trainees and the instructors.

## 2.3 Previewing the Course Exercises

One task that is sometimes overlooked is the training of the instructors. Each instructor should be familiar with the exercise that the trainees will be doing and ideally the instructor should have already done the exercise to understand ahead of time what the trainee will see.

A preview of the exercises and the facility is critical. For the three courses that have been presented, previewing the exercises in the facility was not always possible leading to some very stressful times for the instructors.

During the second PDD course, it was found that a miscommunication between the facility and course organizer resulted in the pond maps laid out incorrectly in the workbook. This made all of the exercises unworkable as they were written. Fortunately this was discovered in a preview the day before the field exercises started and the appropriate changes were made before it affected the course.

During a preview, simulate measurement errors the trainees will make. What is the response if the trainee makes a mistake during the measurement? It is often easier to recover if you've already faced the situation.

The preview also provides the instructors with an opportunity to check the operation of equipment, ensuring that nothing was damaged during shipment and gives the instructors time to correct any problems.

Most importantly, remember that the course isn't going to go exactly as planned. Be prepared to make last minute changes. Sometimes these unexpected events provide great teaching opportunities.

## 2.4 Course Delivery

### 2.4.1 Lectures

The course lectures are designed to provide the trainees with a theoretical background of the principles of Cerenkov light and how Cerenkov light production is used to detect partial defects. This knowledge allows the trainees to understand what is happening when they are looking at partial defects and make decisions based on that knowledge. Additional lectures include the operating principles of the instrument and the characteristics of spent fuel with missing or substituted fuel pins.

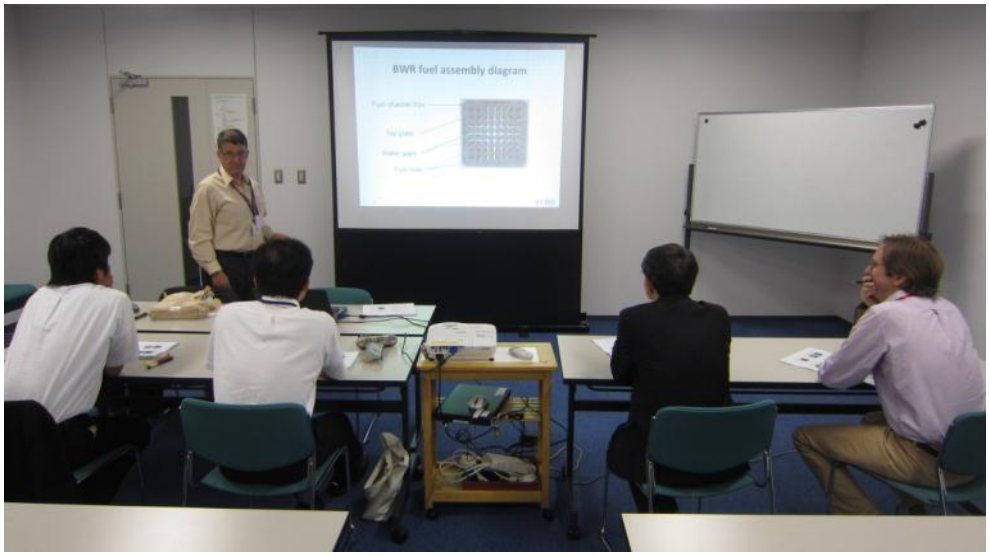


**Figure 2: Course lectures provide the theoretical basis for partial defect detection**

The lectures are continually refined based on feedback from the trainees to provide the information in an interesting and understandable manner. A significant change has been made to the format of the lecture slides, using the “assertion-evidence” design model for scientific presentations[1]. The format of the lecture slides seems to be much more effective during the presentations. Remember: the purpose of the slides is to aid the trainee in understanding the content rather than provide talking points for the instructor.

### 2.4.2 Field exercises

The field exercises are designed to teach the trainees how to use the instrument in a real world situation. This allows the trainees to directly apply the knowledge gained during the lectures The trainees can work without the pressure of an inspection while being supervised and directed by an instructor. To formulate the exercises, the facility hosting the training is asked to provide a pond map indicating the location of the spent fuel assemblies and each assembly’s burnup and cooling time. From this information, a series of exercises is designed to teach the method of partial defect detection.

The exercises start with the basic operation of the DCVD. Once that skill is mastered, the trainees progress to orientation and alignment of the instrument over a fuel assembly. Next the trainees are required to measure and observe the burnup and cooling time of a fuel assembly noting the intensity of the Cerenkov light and the subsequent measurements taken by the DCVD. Once these principles are understood, the trainees begin to look at spent fuel examining the measurement for the characteristics of a partial defect.

**Figure 3: Trainees perform measurement exercises under the supervision of an instructor**

The exercises start at a very basic level and each one builds on the previous exercise. Ensuring that each trainee has enough time to perform the measurement is an administrative challenge and relies heavily on a realistic schedule. Trainees work in groups of two or three so the instructors must be careful to ensure that each trainee has the opportunity to try the exercise and that each exercise is understood.

### 2.4.3 Troubleshooting sub-optimal conditions

One of the benefits of having exercises in a real facility is that the trainees gets to work with the instrument and cope with real world conditions. The delays in getting through security, equipment that is not delivered on time, dealing with real time constraints have all been experienced first-hand. In the case of the DCVD, additional issues can be illustrated including water turbulence, issues with ambient light, and the effect of near neighbours.

## 2.5 Getting Feedback

The preparation of the course started with a clear set of objectives and there should be some mechanism to determine whether they have been met. Two methods of measuring the success of the knowledge transfer in the course are an end of course survey and a course quiz.

The course quiz should have questions related specifically to the objectives. The quiz is not necessarily a pass/fail but an evaluation of how well the material being transferred is understood by the trainees. The results of the quiz should be reviewed with each trainee. If properly developed, the quiz will give a good indication of the level of understanding held by the trainee and also will provide an indication of where the course is deficient in transferring knowledge and developing skills.

A final survey should also be prepared for the course. The trainees should be able to indicate the satisfaction with the training materials, the facilities and how well the objectives of the course were met. The survey should also provide the ability to make suggestions and comments on the training course itself.

# 3 Format of the three previous courses

The partial defects course has been presented three times; each with the same content but with a different format. The time constraints, availability of the inspectors and the facilities were different for each course.

The initial course was held at CLAB the Swedish interim spent fuel repository. This course was held over four days, the first day and a half for security/entrance and lectures and the following two and a half days for in-field exercises. A total of nine inspectors were trained.

The second course was presented in Japan. To accommodate the inspector's schedules, the course was modified to provide a day of lectures and a day of field exercises. A total of twelve inspectors were trained. Each day four inspectors would arrive and would be trained over two days. The lectures were repeated on three consecutive days and the field portion of the course was repeated on three days as well.

The most recent partial defects detection training was integrated into the Spent Fuel Verification course which was held in November 2012 in Ringhals NPP, Sweden. The lectures were integrated into the SFV course which was delivered at the IAEA in Vienna. The field exercises were done during the field portion of the course at Ringhals.

## 3.1 Observations from the courses

Of the three courses that have been presented to date, the instructors and the trainees agree that having the lectures and then the field part of the course over a few days works best. There is a significant amount of material to cover during the course. The compressed version delivered in Japan made it quite difficult to get through all of the material while expecting the trainees to retain the knowledge and skills.

One of the observations that have been made over a number of courses is that the first day in a field training course is usually not entirely productive. The early part of the day is generally filled with entrance security and facility training requirements. There is also a learning curve involved with dressing in protective clothing, getting facility dosimeters, access cards and biometrics not working. All of these factors usually fill half a day or more before the trainees reach the fuel pond (then of course it's lunch time).

Providing the material at a steady pace makes it easier for the trainee to absorb the material. The days in a reactor hall or a spent fuel pond are tiring and it is easy to see the level of energy in the trainees decrease over the day. If possible it is best to have the trainees working on an exercise or solving a problem near the end of the day as little lecture material will be absorbed when the trainees are tired.

It seems that no matter how carefully the lectures are planned, and how detailed the walkthrough of a procedure is done, the trainees learn best when they are forced to perform an activity on their own. If time permits, having a trainee work one on one with an instructor is very valuable. The instructor can more easily asses the trainee's skills and provide specific direction. In any case the groups should be kept small. Two or three people seem to work best. There will always be trainees that minimize their participation and it is important to make sure that everybody spends time working with the equipment.

Holding a wrap-up session at the end of the day in which the trainees can provide feedback on the day's activities and ask questions is a valuable exercise. It is important to make sure that everyone participates, even if it means scheduling certain people or groups to make a presentation at the wrap-up session.

Maintaining the attention of the trainees in these days of smartphones, email and texting is often a challenge. Experience has shown that courses where the lectures are held inside a facility where cell phones are not allowed can be very beneficial. When scheduling activities, remember to plan

something interactive in the time shortly after lunch to help the inspectors (and instructors) stay involved and awake. A long technical lecture shortly after lunch is usually not a good idea.

### 3.2 Dealing with language barriers

The three courses that have been held so far have been delivered in English. For most of the trainees on these courses, English is not their first language. This has to be carefully considered when preparing lecture material and preparing the instructions for the workbooks.

In general, the lecture slides should illustrate the concept using images and animations where possible. Most trainees also find this more interesting and easier to understand as well. The terminology must be familiar or be well explained during the presentations. The pace of the presentations must also be reduced somewhat to allow the trainees to internally translate the material if necessary.

The trainees should be encouraged to ask questions when they arise and the question should be repeated before being answered.

## 4 Recommendations for setting up a “hands on” training course

1. Be realistic about the objectives of the course. Make them measurable.
2. Give a lot of thought to how the course is scheduled. Make the most efficient use of the time in a facility.
3. Be careful how much material will be scheduled for each day. A trainee can only absorb so much in a day.
4. Try mixing lecturing, hands on learning and interactive exercises to keep the trainees engaged.
5. Be prepared for sudden changes to the schedule. Things will always happen to disrupt the schedule.
6. Prepare a course summary report describing general impressions about what worked and what didn’t. This document is invaluable for planning future courses, especially if new course management is involved.

## 5 Future Courses

The next PDD course will be held in June in Sweden and will incorporate many of the things we have learned in previous courses. The course in June will be held using the same model as described in the first course although it is hoped that it can be scheduled so the lectures and practical exercises can be intermixed to increase the effectiveness of the training.

The course will continue to evolve and improve based on the feedback of the trainees and the instructors.

## 6 Acknowledgements

The authors would like to acknowledge the kind cooperation of the CLAB Interim Storage Repository in Sweden, the Hamaoka NPP, Japan, and the staff at the Ringhals NPP in Sweden for their cooperation and enthusiasm in hosting these courses.

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# Safeguards for a Disposal Facility for Spent Nuclear Fuel – the Construction phase

**Mikael Moring, Marko Hämäläinen, Tapani Honkamaa, Elina Martikka, Olli Okko**

STUK – Radiation and Nuclear Safety Authority
P.O. 14, 00881 Helsinki, Finland
E-mail: `firstname.lastname'@stuk.fi

***Abstract:***

*The application for a license to construct a final disposal facility, which will consist of an encapsulation plant and a geological repository for spent nuclear fuel, both in Olkiluoto, was submitted at the end of December 2012 by the implementing company Posiva. STUK will give a full evaluation of the application to the Ministry of Employment and the Economy by mid-2014. After the review the license can be granted and the construction of the first final disposal facility for spent fuel in the world may start in early 2015. As required by Finnish law, a part of the submitted application is a plan for arranging the nuclear materials safeguards, which are necessary to prevent the proliferation of nuclear weapons.*

*A final repository, where any retrieval of the fuel is unlikely to ever happen, sets unique challenges also on safeguards. As reverification is not possible, the whole process has to give full and final assurance that the fuel has been stored according to specifications and that no diversion has been possible. STUK is evaluating Posiva's plan with the aim that Posiva will be able to fulfil all requirements of the Finnish law and of international treaties. As the repository is the first of its kind, it sets new challenges e.g. because the verification requirements of the IAEA and the European Commission are not yet specified. STUK is actively cooperating with the nuclear waste management company Posiva and the international organisations so that all elements of safeguards will be ready and aligned before the start of the nuclear operations at the disposal facility in 2020.*

**Keywords:** safeguards, final disposal, spent nuclear fuel, SbD, 3S

## 1. Introduction

A final disposal facility for spent nuclear fuel is being developed in Finland. The original Decision-in-principle regarding the final disposal of spent fuel produced by the Olkiluoto 1 & 2 and Loviisa 1 &2 units, using the KBS-3 concept [1] was ratified by the Finnish parliament in May 2001.The operator is Posiva Oy, which is owned by the nuclear power companies TVO and Fortum. Posiva has been excavating an underground rock characterisation facility called "Onkalo" in the Olkiluoto island in Eurajoki since 2004, and is thus preparing for the construction of the final disposal facility. While neither a nuclear licence holder nor a nuclear material holder yet, Posiva is foreseen to develop a new type of facility, the geological repository, where the nuclear material cannot be re-verified once it has been encapsulated and emplaced. In the IAEA safeguards approaches it has been suggested that the geological formation should be under safeguards during and beyond the lifetime of the underground facility, or as long as the NPT is in force. Therefore, Posiva's activities in Olkiluoto have been put under national safeguards measures already before Posiva will become a nuclear material holder. The main safeguards activities have aimed at making sure that Onkalo is built as declared and that there are no undisclosed safeguards relevant activities on the site. These procedures are described in a STUK technical report [2]. The preliminary basic technical characteristics (BTC) have been provided and the European Commission has already assigned the material balance area (MBA) code W0LF for Onkalo, and the facility, without nuclear materials but having provided the BTC, constitutes a site according to the Additional Protocol [3]. By the end of 2012, Posiva submitted to the Government an

application to construct the final disposal facility, which will consist of an encapsulation plant and a geological repository. [4]

Finland, as a member state of the EU is a part of Euratom safeguards as well as a sovereign member to the IAEA. The additional protocol (INFCIRC 540) was ratified by Finland in the year 2000 and entered into force in 2004, when all EU member states had ratified it. Nuclear safeguards is a legally binding international instrument, and a prerequisite to all peaceful use of nuclear energy. The scope and mandate for Euratom nuclear safeguards are defined in a European Commission Regulation [5], and they are directly legally binding for each operator.

In Finland, STUK is the competent authority regarding the implementation of nuclear non-proliferation. The national nuclear safeguards derive their mandate and scope from the Finnish Nuclear Energy Act [6] and Nuclear Energy Decree [7]. These were amended during 2008 as a result of the general constitution based renewal of the Finnish nuclear legislation system. An operator's obligation to have a nuclear material accountancy system and the right of STUK to oversee the planning and generation of design information for new facilities was introduced from STUK regulations to the Decree.

## 2. Entering the Construction Phase

The application for a license to construct a final disposal facility, which will consist of an encapsulation plant and a geological repository for spent nuclear fuel, both in Olkiluoto, was submitted to the Ministry of Employment and the Economy at the end of December 2012 by Posiva. In addition to the application, Posiva delivered detailed documentation to STUK. This documentation includes their preliminary safety and probabilistic risk assessments, their proposal for safety classification of structures, systems and components, their quality management during construction, their safeguards, security and emergency plans and their assessment of post closure safety of the facility.

STUK plans to give a full evaluation of the delivered material to the Ministry of Employment and the Economy by mid-2014, presuming that the quality of the material facilitates an effective review. The evaluation is a large undertaking, about 13 person years, in 2013, divided on among 70-90 persons. If based on regulatory review all the safety, security and safeguards requirements are met, a construction license can be granted and the construction of the first final disposal facility for spent fuel in the world may start in early 2015. As required by Finnish law, a part of the submitted application is a plan for arranging the nuclear materials safeguards, which are necessary to prevent the proliferation of nuclear weapons [7].

A formal BTC update, which is based on the same plans and material as the license application, is due to be submitted by Posiva to the Commission by the end of May 2013. Once this BTC update is submitted, also the international inspectorates will have the most recent, fully relevant and official information about Posiva's project plans needed to make detailed plans of their safeguards activities and requirements.

## 3. Safeguards Challenges

According to Posiva's plan [1], the fuel elements will be stored in copper canisters with a welded lid, at the encapsulation plant. In practice the welding of the canister lid marks the point from which fuel elements cannot be easily accessible for verification. In the current application, there are plans for two interim storages for welded canisters, one at the encapsulation plant, and another underground, at the repository level. Containment and Surveillance (C/S) measures need to be developed and implemented to ensure that no diversion of filled canisters can take place. Radiation can be measured from the canisters, but it is uncertain if a radiation finger print can be created that would uniquely identify them. One problem with designing measurements on canisters is that there will be no copper canisters filled with actual nuclear fuel to test on, before the encapsulation facility is built and ready to go into production mode.

For operational and long term safety reasons, the operator intends the volume of open tunnel to be as small as possible. Thus, once a canister is placed into the final position in the bedrock, the emplacement hole is quickly filled up with bentonite clay blocks, and once a disposal tunnel in the

repository is filled with canisters in their final locations, the specific tunnel will quickly be filled up too, and a thick concrete plug will be made at the end of the tunnel [1]. At this point the fuel stored in the backfilled tunnel should be regarded as impossible to access, at least on any foreseeable safeguards grounds. The continuation of knowledge (CoK) must then be based on accountancy and documentation and continued forever.

As the planning proceeds, and Posiva soon will enter the construction phase, and subsequently also the operational phase of the facility in 2022, it is time to get into the more detailed planning of how to safeguard this facility. STUK and Posiva are continually working together with the EC and the IAEA, with a goal to develop SG practices that are both effective and cost effective, but also as unobtrusive on the plant operations as possible. These informal discussions have been fruitful, and are building a common understanding and a way ahead. However, as the work proceeds, there is also a need for more formal commitment to specific procedures.

The current plans include NDA verification of all fuel elements before the encapsulation. Ideally, to facilitate efficient C/S, this final verification measurement should be performed as close to the encapsulation as possible e.g. at the encapsulation plant. This goes against previous suggestions [8,9] that the verification should be performed in an early phase, to give the inspectorates time to fully resolve any possible inconsistencies. C/S measures of the encapsulation plant should be possible using standard safeguards instrumentation as portal monitors, and surveillance cameras. The data from the measurement equipment should be shared between all involved parties, giving everyone time to make their necessary conclusion within the very tight timeframe given.

C/S for the underground repository, however, includes new challenges, as the fuel will eventually become impossible to access for reverification. The safeguards measures to be used have not been decided upon yet, but at a minimum they should include radiation monitors at all the entrances (tunnel and shafts) through which nuclear material could be transported out of the facility.

In order to support accountancy, the canisters will be provided with an engraved (or relief) serial number, from which they can be uniquely identified until the point when they are finally stored and the emplacement holes are backfilled. Through this number, accountancy data can be checked all the way until the canister is stored in its final position and the hole is backfilled. There are ongoing discussions on more elaborate fingerprinting methods, e.g. based on individual weld inspection data of the canisters, but no such method has yet been proven viable or decided upon. Additional methods that can be used underground are short notice inspections and environmental sampling, to ensure that no undeclared activities are taking place below the ground.

## 4. Safeguards in Design and Construction

The importance of including safeguards already in the design phase is heightened, as the disposal facility will be a completely new form of nuclear installation, with unique safeguards challenges. This requires real effort not only from the operator and the national authorities, but also from the international inspectorates. The facilities (encapsulation plant and final repository) need to be planned and built in a diversion resistant manner which facilitates cost effective safeguards. The physical extent of the geological repository will change during the operational period that is longer than 100 years. Design of the underground facility is a continuous process based on geological findings. Therefore, the safeguards measures have to evolve and be adopted accordingly during the construction and operation of the facility.

One successful example of the ongoing cooperation already in the design phase is the inclusion of positions for non destructive assay (NDA) measurements of the fuel elements in the hot chamber of the encapsulation plant. Such positions did not previously exist in Posiva's blueprints for the encapsulation plant, although the need was mentioned at least as early as in 1999 [10]. Adding them to an already constructed hot chamber might have become impossible, or at least very costly.

## 5. Safety, Security and Safeguards (SSS)

The final disposal facility is a pilot project also for developing the three-S concept at STUK. The synergies and conflicts between the safety, security and safeguards requirements need to be handled in a cooperative way, to enable deeper understanding of areas where specific requirements have effects on requirements in one or both other areas. STUK has started an interdisciplinary project between the sections within STUK, working on these areas. In utilizing synergies it is often possible to avoid double work, by using practices demanded by another process. For example, safety, security and safeguards can all benefit from the same NDA measurement, even though the focuses differ. Security needs to ensure that no nuclear matter has been diverted, Safety needs to know that there is no more nuclear material in the canister than declared (because of heat loading concerns) and Safeguards needs to verify that the correct declared amount of material is in the canister. The thorough monitoring of the safety aspects of the final repository gives added confidence also in the security and safeguardability aspects.

The probably best known conflict lies between occupational safety and security. Where personnel safety needs open passages in case of an emergency, top level security requires tight control of doors, areas and people's movements. A conflict has also been identified between safeguards and security, e.g. the remote operation of safeguards equipment raises valid concerns about data and ultimately operational security of the plant. When such rivalries arise, it is imperative that the issues are handled by the relevant persons in cooperation, to find the most constructive paths forward.

## 6. Conclusion

The construction of the first disposal facility for spent nuclear fuel in the world will, very likely, start in 2015, with operation planned to start in 2022. There is a fruitful cooperation going on between the IAEA, the European Commission, STUK and the operator Posiva. However, time is scarce, and development of approved new procedures, which might even involve the development of new technology, is a very slow process. The time to act is now in order to have the safeguards precautions included in the nuclear construction licence that has already been applied for.

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# Numerical Modelling of Corrections relevant to Nuclear Accountancy Measurements by Hybrid K-edge / XRF Densitometry

**M. Toma[1,2], P. van Belle[1], K. Lützenkirchen[1], D.A. Sharikov[3]**

[1]Institute for Transuranium Elements
Joint Research Centre, European Commission
PO Box 2340, 76125 Karlsruhe, Germany
E-mail: magdalena.toma@ec.europa.eu,
[2]Horia Hulubei National Institute of Physics and Nuclear Engineering - IFIN HH
Str. Reactorului no.30, P.O.BOX MG-6, Bucharest - Magurele, ROMANIA
[3]Tomsk Polytechnic University, 30, Lenin Avenue, 634050, Tomsk, Russia
E-mail: sharikov@tpu.ru

***Abstract:***

*Hybrid K-edge densitometry (HKED) is a combined x-ray transmission and x-ray fluorescence spectrometry technique for the non-destructive assay of U and Pu in different types of Nuclear Safeguards samples such as those encountered in existing spent-fuel Reprocessing Plants. A similar Hybrid K-Edge instrument is ready to be installed at an ITU hot cell for measurements on samples from pyro-chemical, as opposed to conventional PUREX, reprocessing of spent fuel.*
*Unlike the low-Z matrices of samples encountered in traditional reprocessing, the matrices of samples from pyro-reprocessing may contain high-Z elements, such as Cd or Bi.*
*To reduce the calibration effort, which is normally needed for accurate routine HKED measurements, and to predict the response of the instrument for non-standard samples, e.g. U-Pu mixtures with unusual element ratios or those with high-Z matrices, a Monte Carlo modelling approach was applied. As the excitation of U and Pu is affected by different regions of the incident x-ray spectrum, special attention was paid to reconstruct accurately the spectrum of the incident x-rays from experimental measurements. In routine measurement of the U/Pu ratio, samples with similar but slightly different composition or density are analysed. A study of the effect of a sample's composition/density variations on experimentally observed U/Pu fluorescence peak area ratios has been undertaken. The results of the study are reported in this paper.*

**Keywords:** Nuclear Safeguards; HKED; Monte Carlo modelling; calibration;

## 1. Introduction

The reprocessing of spent nuclear fuel allows both uranium (U) and plutonium (Pu) to be fully separated from fission products. The U and Pu product might thereafter be used to produce fuel for MOX-fuelled reactors, while the separation of the fission products greatly reduces the volume of nuclear waste that needs to go to final repository. As the Pu product could potentially be used for non-peaceful purposes, Safeguard measures are in place at the Reprocessing Plants to prevent, or to detect in timely manner, the illegal diversion of nuclear material, e.g. for nuclear weapon production.

Hybrid K-edge / XRF densitometry (HKED) has become an important tool for Nuclear Safeguards at the European Reprocessing Plants of La-Hague (France) and Sellafield (Great-Britain), which are both under EURATOM Safeguards control. The Japanese Safeguards Authority (NMCC) together with the IAEA operate similar instruments at the Japanese Reprocessing Plant at Rokkasho. The HKED is the instrument of choice as it offers a convenient, fast and accurate means to determine the U and Pu content of process solutions from these Reprocessing Plants. Further HKED system are in use in the ITU (Karlsruhe, Germany) for routine sample analysis, for the improvement of the HKED methodology

at the Reprocessing Plants and for the development of HKED measurement methods on samples arising from the more advanced pyro-chemical reprocessing of spent fuel.

The principle of a Hybrid K-edge/XRF measurement is simple [1]. A sample held within an appropriate sample container[1] is placed within the HKED measurement geometry where it is exposed to x-radiation from a 150 kV x-ray generator. The incident radiation from the x-ray tube is sufficiently energetic to remove the strongly bound K-electrons from the uranium and/or plutonium atoms that are present in those samples. The removal of the K-electron is associated with a sharp increase in attenuation cross-section, and the attenuation itself depends on the U and/or Pu concentration of the sample. A High Purity Germanium (HPGe) detector is used to measure the x-ray energy spectrum transmitted through the sample, so that the change in sample attenuation at the energy of the K-edge (115.61 and 121.80 keV for U and Pu respectively) can be quantified. A calibrated K-edge instrument can then convert the observed change in sample attenuation into a nuclear element concentration.

In practice, the K-edge transmission measurement only yields accurate data for the actinide with the highest concentration. Reprocessing input solutions typically have a U/Pu atom ratio of about 100 and typical actinide concentrations of 200 gU/L and 2 gPu/L. Dissolved MOX samples typically have a U/Pu ratio of about 10 with actinide concentrations of about 160 gU/L and 16 gPu/L. In input solutions and dissolved MOX, uranium is the major element and is accurately measured by K-edge. By contrast, the plutonium concentration in these samples is insufficient for assay by K-edge and is determined instead by measuring the U/Pu atom ratio by x-ray fluorescence (XRF). Actinides that have lost a K-electron are electronically unstable and produce fluorescent x-rays with energies that are characteristic of the element. The main K-XRF emission for uranium, U-Kα1, occurs at 98.43 keV, while Pu-Kα1 occurs at 103.74 keV. The intensity of both the U-Kα1 and Pu-Kα1 emissions are measured by a second HPGe detector observing the sample at an angle that is 150 degrees displaced from the angle of the K-edge transmission measurement. After calibration, the XRF measurement yields an absolute U/Pu atom ratio. In the Hybrid K-edge/XRF instrument the absolute assay of the uranium atom concentration by K-edge is then combined with the XRF measurement of the sample's U/Pu atom ratio to yield the absolute plutonium atom concentration.

The K-edge is further used to determine the amount of U in concentrated but pure uranium samples (e.g. uranyl nitrate or dissolved U-oxide samples) or the amount of Pu in pure Pu product (e.g. plutonium nitrates or dissolved Pu-oxide). Samples with very low actinide concentrations can not be determined by K-edge and are measured instead using an absolute measurement of the U-Kα1 and/or Pu-Kα1 emission by XRF.

Although HKED measurement principles are straightforward, there are effects that need correction before accurate and unbiased results can be reported. The most important amongst these effects is the fact that both the K-edge excitation and the escape probability of the various x-ray emissions are dependent on sample density and heavy element concentration. In the U/Pu measurement by XRF, the more energetic Pu-Kα1 x-ray is marginally less affected by attenuation than the lower energy U-Kα1 emission, leading to a measured U/Pu response that also depends among others on heavy element concentration. Traditionally, the influence of such effects would be automatically absorbed when calibrating the HKED instrument with a set of certified calibration solutions that adequately cover the expected range of uranium concentrations and U/Pu ratios. However, such multi-point calibrations are expensive, time consuming and render the instrument unavailable for routine sample measurement while recalibration is in progress. If preparation of all calibration solutions is included, a full recalibration of the HKED instrument for PUREX input solutions requires about eight to ten weeks of manpower, with serious impact on the laboratory's sample-throughput. Calibrating the instrument for MOX requires similar effort.

Considering the major impact that instrument calibration has on laboratory productivity, it is evident that the ITU is looking into alternatives to the expensive multi-point approach. The most promising approach is that of a single-point instrument calibration augmented by accurate knowledge, obtained by MCNP [2] photon transport modelling, of effects such as the influence of heavy element and matrix concentration on the measured U/Pu response. A single-point calibration involving a physical sample side-steps the need for MCNP modelling to be absolutely accurate to Safeguards accuracy (about

[1] There are currently two types of container that are in common use: the ITU container, used in ITU and Sellafield, with a diameter of 9 mm for the XRF cell, and the COGEMA container, used in La-Hague and Rokkasho, with a 14 mm diameter.

0.25 % 1-s) and only requires modelling to deliver modestly accurate correction factors. Such corrections, while essential, are comparatively benign and the limited accuracy of modelled correction coefficients will therefore only have an acceptably small impact on the overall uncertainty budget for the final assay results. Correction coefficients derived from multi-point calibrations are not free from uncertainty either and the indications are that a single-point calibration augmented by modelling can deliver the same performance as an expensive multi-point calibration. The modelling approach, once validated, can then also be deployed in situations where the use of multiple calibration solutions becomes impractical or is prohibitively expensive.

The MCNP photon transport modelling code is perfectly capable of dealing with the simulation of the photon transport of our HKED/sample geometry. However, the results of the simulation are only expected to be reliable when care is taken to set up the model correctly. A critical aspect is the radiation source-term of the model, i.e. the description of the x-ray intensity spectrum emitted by the x-ray tube. The manner in which this source-term was obtained from experimental data is described in section 2. To obtain statistically significant modelling results, the MCNP code needs to compute the fate of billions of photons which, even on today's powerful Personal Computers, still takes about one day per simulation. To enhance our understanding of photon transport within the HKED geometry and to obtain initial estimates of the magnitude of various correction factors for anticipated sample characteristics, we anticipated that hundreds of those simulations would be needed. MNCP offers several so-called variance reduction techniques to speed up its simulations, but, as such "short-cuts" may affect the reliability of the final result, effort was put into the cross-comparison of the various computational expedients offered by MCNP. This study is described in section 3. Having settled on a mode of computation, our MCNP models were then exploited to obtain the main correction factors for cases that are relevant to our current measurement efforts, i.e. our measurements on samples from traditional PUREX-based reprocessing of spent fuel, as well as for future efforts, e.g. the anticipated measurements on samples from advanced reprocessing by pyro-metallurgical separation. In contrast to samples for and from the PUREX process where the main sample matrix is aqueous nitric acid, the samples arising from pyro-reprocessing will have matrices that contain significant quantities of moderately-heavy to heavy elements such as cadmium and bismuth. Modelling of the latter class of samples is useful in providing early indications of the problems that will be encountered during the measurement of pyro-samples and provide an insight into the magnitude of the corrections needed for these exotic samples. The deployment of the modelling to some selected but representative PUREX and pyro-samples is described in section 4.

## 2. Determination of the incident spectrum of the x-ray source

The numerical modelling of the behaviour of the HKED instrument is performed using the Monte Carlo N-Particle Transport Code (MCNP) [2]. The general models for the XRF branch of the instrument for both the 9 mm and 14 mm sample containers were previously developed [3]. However, in this section we report on the manner in which better modelling of the incident x-ray source spectrum has been achieved. This is considered important because U and Pu are sensitive to different regions of the incident spectrum (Fig 1) and an accurate prediction of the Pu x-ray detector response relative to that for the U x-rays, needs the best possible construction of the shape of the x-ray spectrum that enters the pre-collimator between the x-ray tube and the sample. Note however, that the model only requires an accurate description of the spectrum for energies above the energy of the uranium K-edge (115.61 keV) because x-rays below that energy can not cause K-ionisation in either U or Pu. This irrelevance of the low energy part of the spectrum also permits the use of a stainless steel plug at the entrance of the pre-collimator to prevent useless events from needlessly burdening the detection electronics.

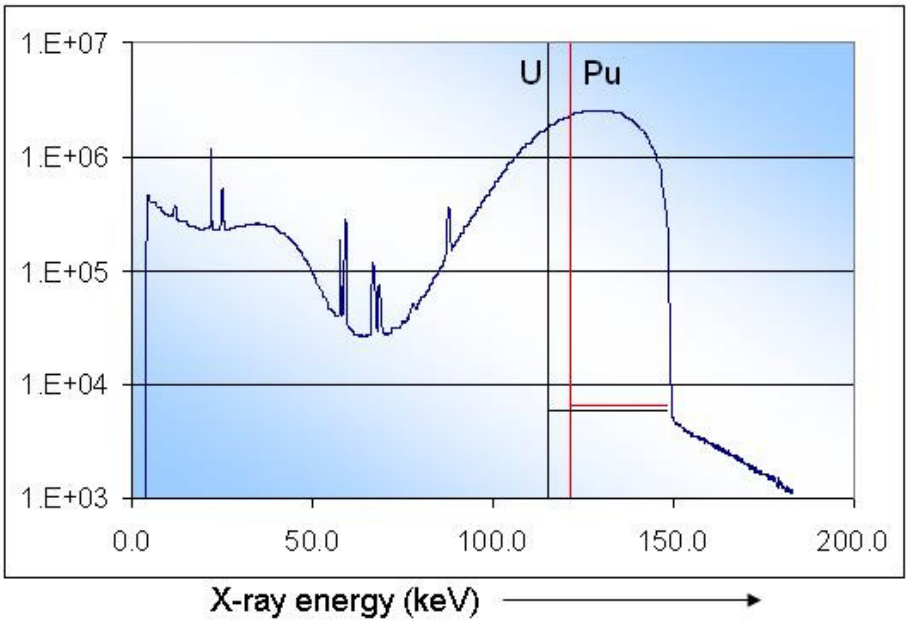


**Figure 1.** Measured K-edge spectrum when no sample is present between x-ray tube and detector. The lines labelled U and Pu mark the lowest energy of the x-ray spectrum causing K-excitation of that element.

Rather than rely on the pure modelling of the incident x-ray spectrum, it was decided to use measured data to construct that source-term. For the most direct measurement of that spectrum, the K-edge detector was used without any sample present in the measurement geometry. The absence of an attenuating sample causes a high count rate on the K-edge detector, so the x-ray tube was operated at its lowest power setting (150.00 kV and 0.10 mA) and the length of the stainless steel plug has been slightly increased to 15 mm to keep the detector count rate at a manageable 13,000 counts/s. The recorded x-ray spectrum still exhibits a modest amount of pile-up (the tail above 150.00 keV seen in Fig. 1), but that pile-up was numerically corrected for before reconstruction of the x-ray source-term. It should be appreciated here that the experimental spectrum is a convolution of the original source-term, the modifying effect of the lengthened stainless steel filter and the energy-dependent detection efficiency of the K-edge detector. Extraction of the pure source-term therefore requires de-convolution of the experimental data for the influence of both the steel filter and the detection efficiency. To obtain data for these convolution terms, the energy-dependent detector response matrix was calculated using a detailed model of the KED source-detector geometry. This model includes the planar HPGe detector with an area of 200 $mm^2$, a thickness 10 mm and a window thickness of 0.15 mm, the tungsten collimator with its 0.8 mm diameter, the 15 mm long stainless steel filter, and has the x-ray tube situated on the detector-collimator axis at 90 mm distance from the collimator (Fig. 2). Using this model the full energy-dependent detection response matrix was then obtained by performing individual transport calculations for a large number of mono-energetic x-ray energies, covering the interval from 98.5—160.5 keV in steps of 1.0 keV. To economise on computation time, the mono-energetic source x-rays, emitted from the 1 mm diameter "spot" on the tube's x-ray producing target, were biased to emit only into a restricted solid angle. That angle is nonetheless large enough to cover the surface of the stainless steel filter as well as some of the surrounding collimator block, so that x-rays that scatter in the tungsten collimator towards the detector are also modelled.

To compute the response matrix, the incident "mono-energetic" radiation is uniformly sampled from a rectangular distribution with a width of 1 keV and is recorded on a grid covering the interval from 0—160 keV in 1 keV bins using the F6 tally of MCNPX (energy deposition averaged over a cell). Having obtained the response matrix as described above, the source-term is then extracted from the pile-up corrected experimental data by means of matrix inversion and spectral stripping. The radiation source-term, so computed, thereafter defines the source in all subsequent models. The validity was checked by using the source-term to predict the experimental spectrum for the situation in which no sample is present in the HKED instrument. That prediction uses the F8 tally of MCNP (energy

distribution of pulses created in a detector by radiation) and Fig. 3. illustrates the acceptable agreement between the simulated and the measured detector spectrum.

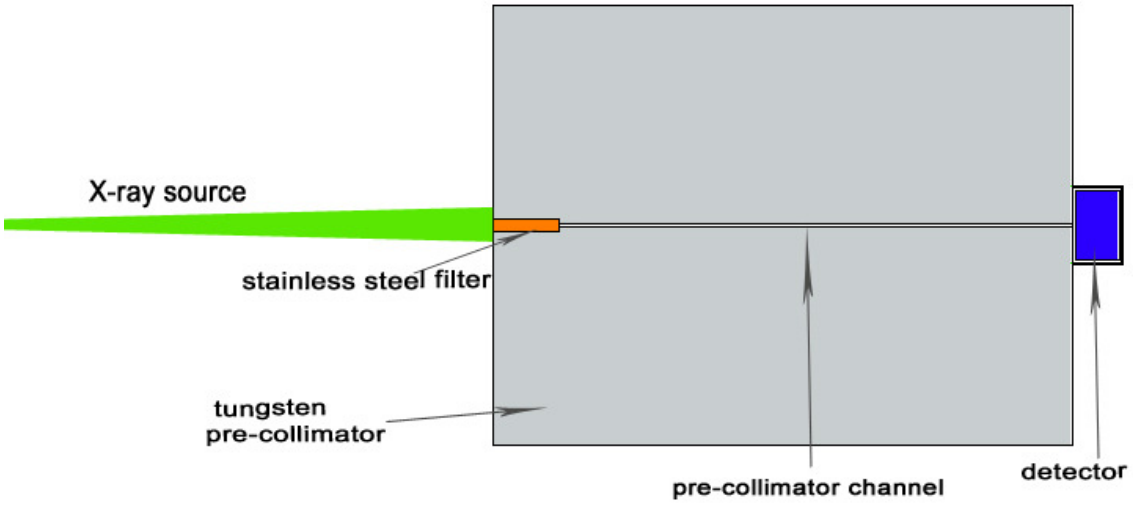


**Figure 2.** Model of the KED source-detector geometry for the purpose of extracting the radiation source-term

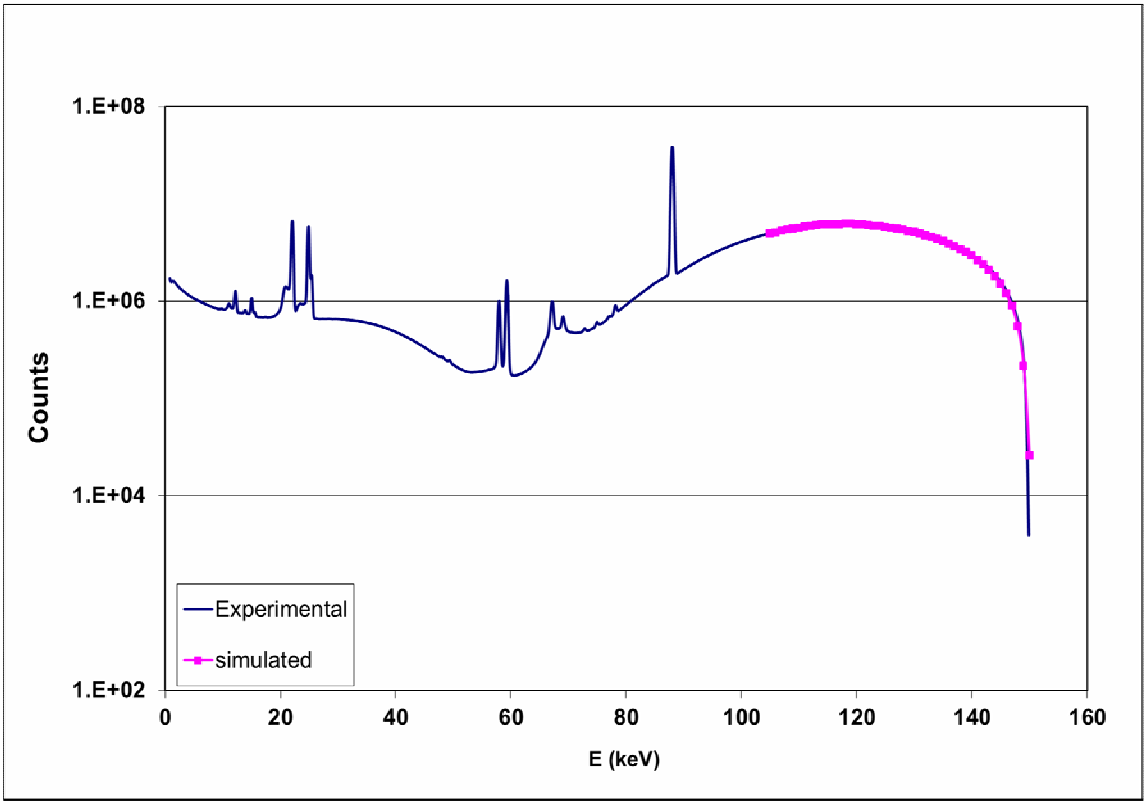


**Figure 3.** Comparison of the measured spectrum and its simulation.

## 3. Study of the impact of using MCNP variance reduction techniques

The need for variance reduction to speed up calculations arises from the computational resources required for high-precision Monte Carlo simulations. The following expedients, available in MCNPX, were studied: the *Point detector*[2] and the *Point detector* in combination with *Forced collisions*[3]. The results are compared to those obtained by using MCNP's *Analogue* mode, a mode that is computationally the slowest, but one that is generally accepted as being the most reliable and realistic.

To assess the impact of the choice of computation mode, a set of aqueous sample solutions with an acidity of 1 molar, a U/Pu ratio of 100 and U concentrations varying from 75—250 gU/L, were studied. Samples of that description are typical for the input samples of the PUREX process and are routinely measured in the EURATOM Safeguards laboratories at La-Hague and Sellafield. The densities of the samples are taken from the SAKURAI and TACHIMORI formula [4] for aqueous solutions containing Pu (IV), U (VI) and nitric acid. The density of the samples varies from 1.136—1.376 g/mL. The simulations are performed for the 14 mm diameter sample containers, because the photon attenuation in these containers is more pronounced than in the 9 mm diameter vials.

**Table 1**. Listing the characteristics of MCNP's main variance reduction strategies

| Calculation Mode | Main Characteristic |
|---|---|
| I | *Analogue*; slow, but considered the most reliable and realistic |
| II | *Point Detector (PD)* only |
| III | *Point Detector (PD)* combined with *Forced Collisions (FCL)* |

For the purpose of the comparison, detector peak areas for U and Pu Kα1 x-rays were computed for all three computational modes listed in Table 1. The analysis of the results shown in Fig. 6, section 4.1, indicates that for all three cases and for the same number of incident photons, the dominant uncertainty of the Monte Carlo simulation is in the computed Pu peak area, with the uncertainty worsening with reducing Pu concentration in the sample. For the same number of incident photons, mode I and II produce results with comparable computational precision, but the precision for mode III is significantly better. This implies, if results need to be obtained to a given precision, that the mode of calculation that combines the *Point detector* with *Forced collisions* will achieve its results significantly faster. The relative time needed to achieve the same precision is illustrated in Fig. 4 and shows mode III to be approximately 2.7 times faster than the *Analogue* transport mode. A calculation that would take three days to complete in mode I takes about one day in mode III.

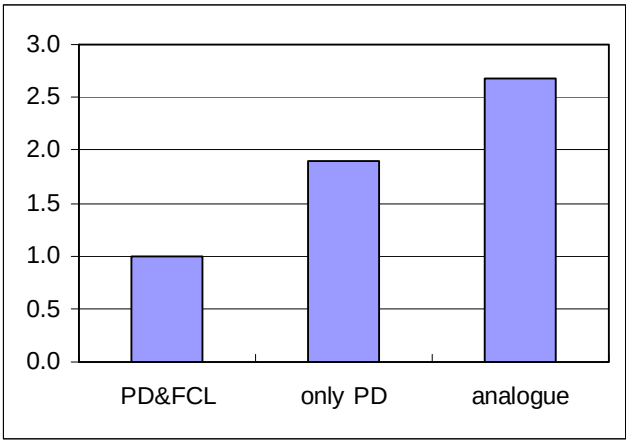


**Figure 4.** The relative computation time for the calculation modes listed in Table 1.

[2] *Point detector (PD):* all interactions in the sample volume contribute to the flux that is emitted towards the detector, ensuring that no x-ray that has been generated is lost for detection.
[3] *Forced collisions (FCL)* in the sample cell: all the photons that reach the sample volume are forced to interact.

The different calculation modes yield slightly different correction coefficients, and, in an approach that combines an experimental single-point instrument calibration with numerically derived corrections for the effects of attenuation for different sample characteristics, the arbitrary choice of calculation mode will slightly affect the magnitude of the correction for samples with significantly different characteristics than those of the calibration sample. In other words, the choice of calculation mode affects the magnitude of the correction and will therefore result in a slight bias in the final measurement result that will depend on that choice. Fortunately, for the realistic scenarios studied, this arbitrary bias is tolerable and appears to be less than 0.2% for the most extreme cases. More detail can be found in the next section.

## 4. Study of corrections for different sample matrices.

### 4.1 Main correction factor for samples that are input to the PUREX process

When assaying dissolver input solutions for PUREX processing, the main purpose of the K-edge detection branch is to provide the absolute measurement of the sample's uranium concentration. This leaves the XRF branch to determine the sample's U/Pu atom ratio, so that, by combining the K-edge and XRF result, the absolute plutonium concentration also becomes known. However, while an accurate measurement by K-edge involves only benign corrections, the same is not the case for the XRF's U/Pu measurement. There are two main effects that influence the U/Pu detection response: 1) the consequences of the fact that different fractions of the incident photon spectrum are used for U and Pu excitation, and 2) the energy dependence of the attenuation of incident x-radiation and escaping fluorescent x-rays. The uranium x-rays have a slightly lower energy and are more attenuated by the sample itself than the higher energy plutonium x-rays. The latter effect becomes progressively more significant with increasing heavy element concentration and matrix density.

The influence of the incident photon spectrum is mitigated by physical control measures that aim to ensure that the energy spectrum of the tube's x-ray emission is kept constant with respect to the spectrum used during instrument calibration. Frequent QA and QC measurements are performed to keep the uppermost energy of the tube emission within 100 eV of the nominal 150,000 eV. Further QA measures are in place that will demand instrument re-calibration whenever the response to standard samples deviates from expectation.

To quantify the main consequence that variations in sample composition have on the measured U/Pu response, a detailed set of calculations were performed for the 14 mm diameter sample geometry, relevant to La-Hague and Rokkasho, and for all three calculation modes listed in Table 1.

It is evident that for samples with a constant U/Pu ratio, the detected peak area for the U-Kα1 and the Pu-Kα1 lines increase with increasing uranium concentration. An example of this is shown in Fig. 5 for the calculations using mode III, but mode I and mode II exhibit similar behaviour.

Purely for the purpose of the following discussion, we pragmatically define *detectability* as the number of U-Kα1 x-rays (or Pu-Kα1 x-rays, depending on the case) that have been detected by the end of a measurement, but both the uranium and the plutonium counts are always normalised to the uranium concentration in gU/L. As attenuation of a sample increases with increasing uranium concentration, the *detectability* of the U-Kα1 x-rays, emitted by simulated PUREX input samples with a modelled U/Pu atom ratio of 100, reduces by a factor of ≈1.62 when comparing a 75 gU/L sample to a 250 gU/L sample. The *detectability* of the Pu-Kα1 x-rays is also affected by the increase in uranium concentration, but here the corresponding reduction factor amounts to ≈1.56. This illustrates that attenuation strongly affects the recorded Kα1 peak area for both elements, but that the U-Kα1 counts are marginally more affected by variations in uranium concentration than are the Pu-Kα1 counts. It further illustrates that it would be very difficult to obtain accurate element concentrations by using the XRF measurement data in an absolute sense: the associated correction factors are simply too large. However, in our HKED application we only require knowledge on the U/Pu atom ratio, and, as the U Kα1 and Pu Kα1 *detectability* are almost equally affected, the measured U/Pu peak area ratio is far less sensitive to uranium concentration changes than the individual ratio components. In fact, for a variation from ≈75—250 gU/L, the U/Pu peak area ratio is predicted to change by a factor of ≈1.035. This change, while far less than the change in the U/Pu's denominator and numerator, is nonetheless

large and requires reasonably accurate knowledge of the magnitude of that correction if the measured U/Pu peak area ratio is to be interpreted correctly.

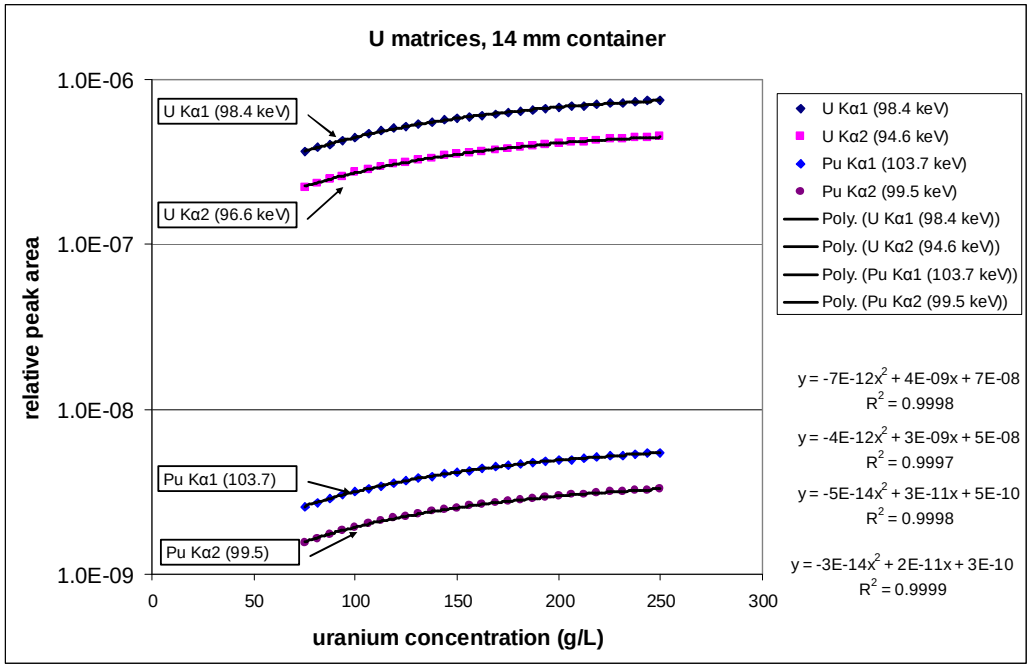


**Figure 5.** Showing the increase in relevant peak areas versus uranium concentration for the 14 mm sample container geometry. The U/Pu atom ratio is kept at 100 for all cases.

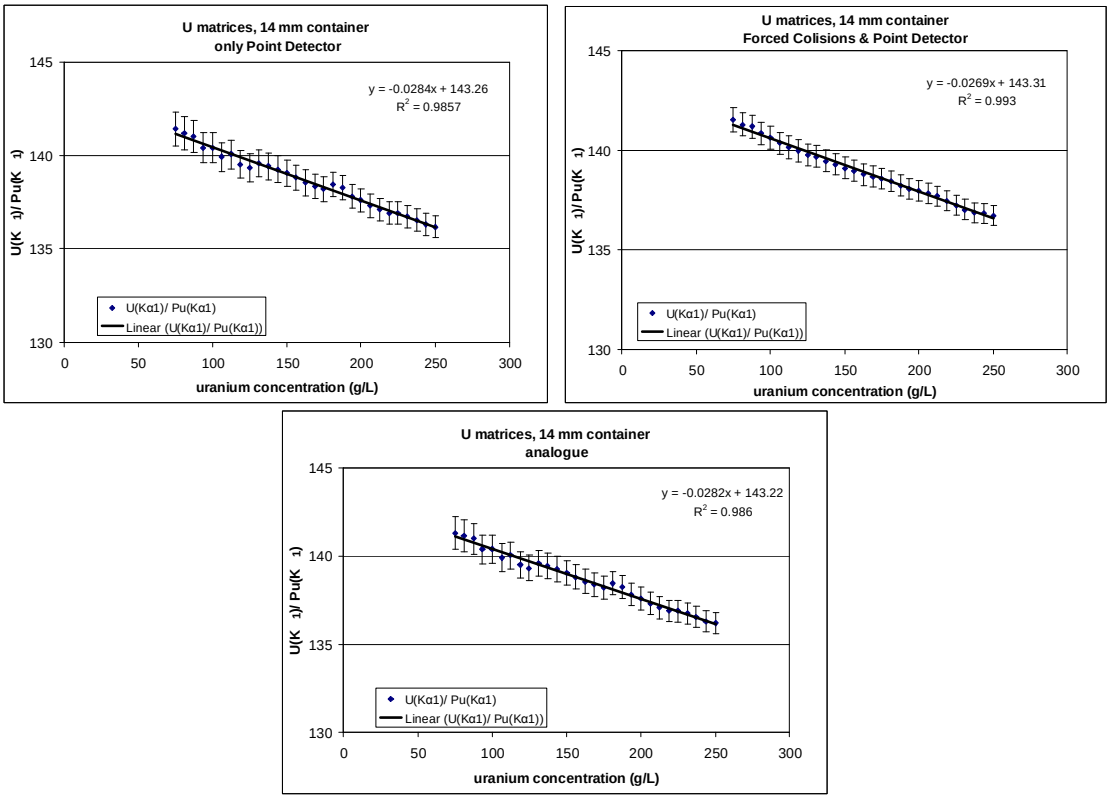


**Figure 6.** Variation of the computed U/Pu peak area ratio with respect to uranium concentration for the three modes of using MCNP listed in Table 1. The U/Pu atom ratio for all calculations is taken as 100.

Given that only the effect of attenuation on the U/Pu ratio is of practical importance, the same range of uranium concentrations for the same range of sample characteristics was used to model the behaviour of the U/Pu ratio using all of the three calculation modes of Table 1. The results are shown in Fig. 6.

All modes of calculation suggest that, if the attenuation effect is not corrected for, the measured U/Pu peak area ratio will yield final results for the U/Pu atom ratio that are biased by up to ≈3.5% across the indicated concentration range.

Fig. 6 further shows that the concentration-dependent correction polynomials, suggested by the three computational models, differ slightly in magnitude. It should be noted at this stage that the high, but limited, precision of the MCNP calculations only permits the most important correction coefficient, i.e. the linear dependence of the correction polynomial, to be determined to limited accuracy. Chi-squared analysis of the MCNP data and its polynomial fit suggests that, for the limited precision of the MCNP calculations indicated as error bars in Fig. 6, the numerical value of the linear dependence in the polynomial can only be determined to approximately 0.20% (1-s). In a worst case scenario, the HKED instrument would be experimentally point-calibrated at 250 gU/L, in which case the most pronounced correction would apply to samples with a low uranium concentration of, say, 75 gU/L. For that extreme difference of 250 - 75 = 175 gU/L, calculation modes I, II and III suggest a correction factor with respect to the point-calibration at 250 gU/L of 1.036, 1.037 and 1.034 respectively. Depending on the choice of model this causes a bias in the measurement of the U/Pu atom ratio of potentially ≈0.25%. However, if the expected variation in the uranium concentration of the routine PUREX input samples would indeed range from 75—250 gU/L, then it is more sensible to perform a physical instrument point-calibration somewhere in the middle of that range. In that case, none of the expected samples deviates by more than 90 gU/L from the calibration-point and the correction factor, which needs to be applied to the most unusual samples, then amounts to no more than ≈1.017 and might introduce a tolerable absolute bias for element ratio determinations by XRF of potentially 0.13% worst case. Finally, it should be noted that these corrections, as well as their potential bias, vanish when the characteristics of a measurement sample approaches the characteristics of the calibration solution that was used in the point-calibration.

### 4.2 Main correction factor for samples from pyro-processing of spent fuel

Unlike the low-Z matrices of PUREX reprocessing samples, the matrices of pyro-reprocessing samples contain high-Z elements such as Cd or Bi. As such matrices are potentially very attenuating, calculations were performed to predict the behaviour of U and Pu peak areas when Bi or Cd is present in the sample. Research into pyro-reprocessing is an ongoing effort at ITU and a dedicated HKED instrument has been built to support this research. We expect that only small sample volumes will be made available for measurement, so the calculations were performed for the 9 mm diameter, rather than the 14 mm diameter, sample containers. The reduced diameter will also mitigate the effects of attenuation on incident source x-radiation and escaping fluorescent x-rays.

The following samples were modelled using computational mode III:

- a set of aqueous nitric acid samples containing Bi, with concentrations varying from 0—300 gBi/L, and all containing the same uranium concentration of 10 gU/L with a U/Pu atom ratio of 100. The densities and elemental composition of the samples were calculated using experimental data from previously prepared synthetic samples. The sample densities vary from 1.227—1.497 g/mL for bismuth concentrations from 0—300 gBi/L.
- a similar set of samples as above, but with Cd instead of Bi and covering concentrations from 0—300 gCd/L. As before, the uranium concentration is kept constant at 10 gU/L with a U/Pu atom ratio of 100. The sample densities and compositions were calculated using existing experimental data. The densities vary from 1.106—1.550 g/mL.
- a set of aqueous sample solutions containing only dilute nitric acid, U and Pu, with a U/Pu atom ratio of 100 and with uranium concentrations varying from 10—300 gU/L. The densities of these samples are calculated using the SAKURAI and TACHIMORI formula [4] and vary from 1.046—1.443 g/mL. This set of samples is similar to the set, described in section 4.1, which was used in that section to obtain corrections for PUREX input samples measured in 14 mm, as opposed to 9 mm, container geometry.

For the matrices containing only U and Pu and nitric acid, the peak areas increase with increasing uranium concentration, but show the expected saturation due to increasing attenuation. The results are show in Fig 7.

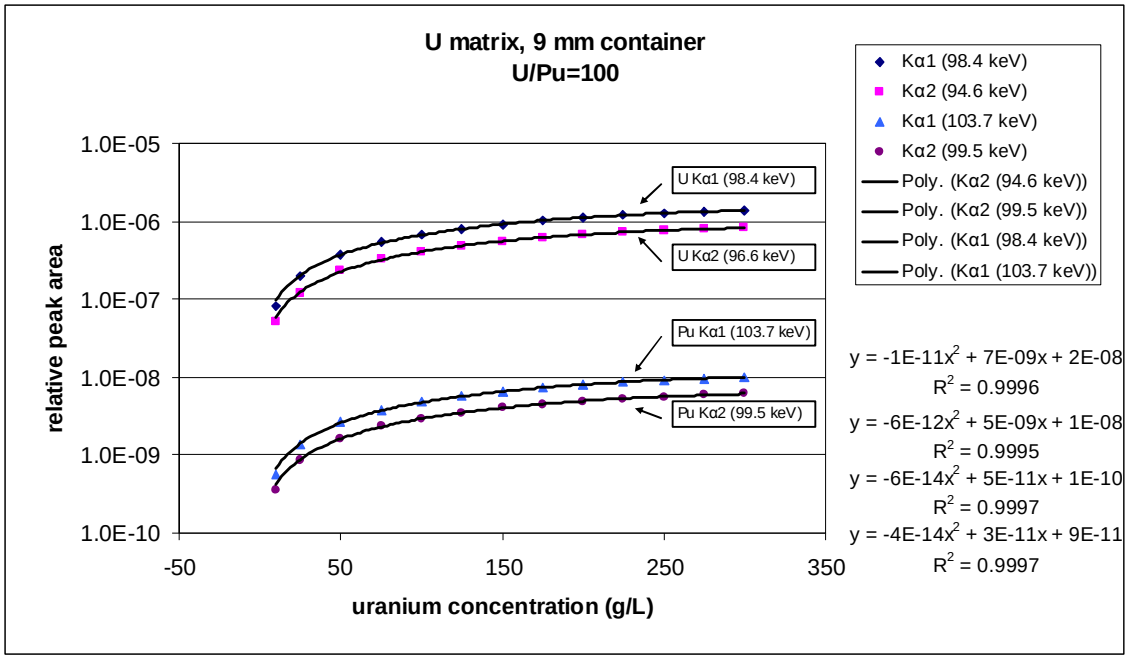


**Figure 7**. Relative peak areas for the samples containing only U and Pu for the 9 mm container

For the bismuth and cadmium matrices, the concentration of U and Pu is taken to be constant for all samples at 10 gU/L with a U/Pu atom ratio of 100. The predicted U and Pu peak areas therefore monotonically decrease with increasing bismuth or cadmium concentration (Fig 8a and 8b).

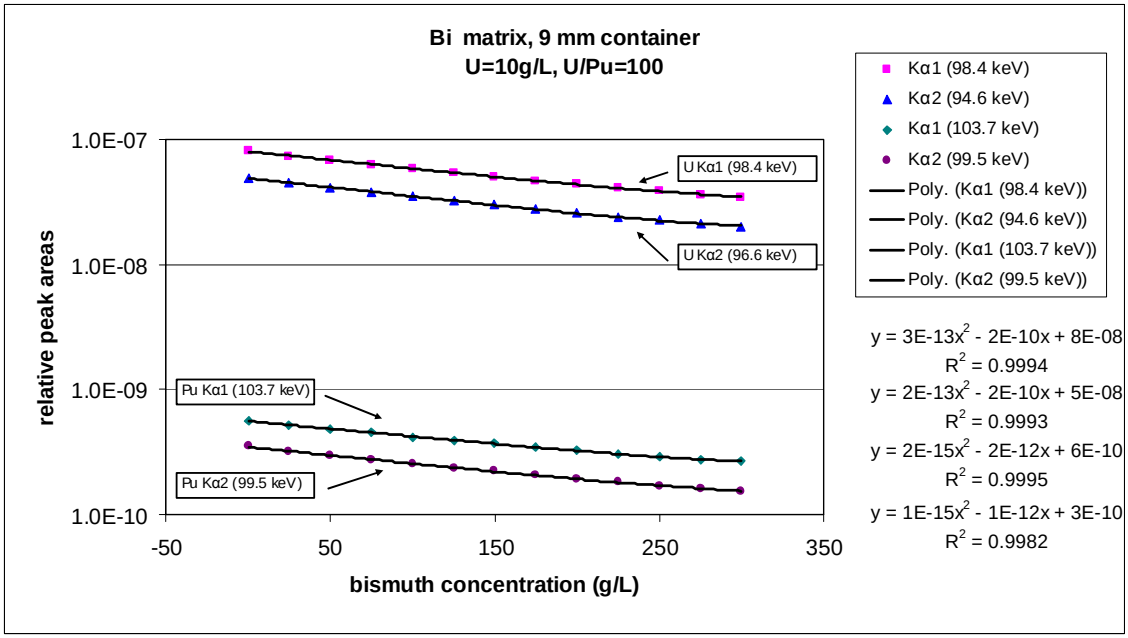


**Figure 8a.** Relative U and Pu peak areas for bismuth samples in a 9 mm container

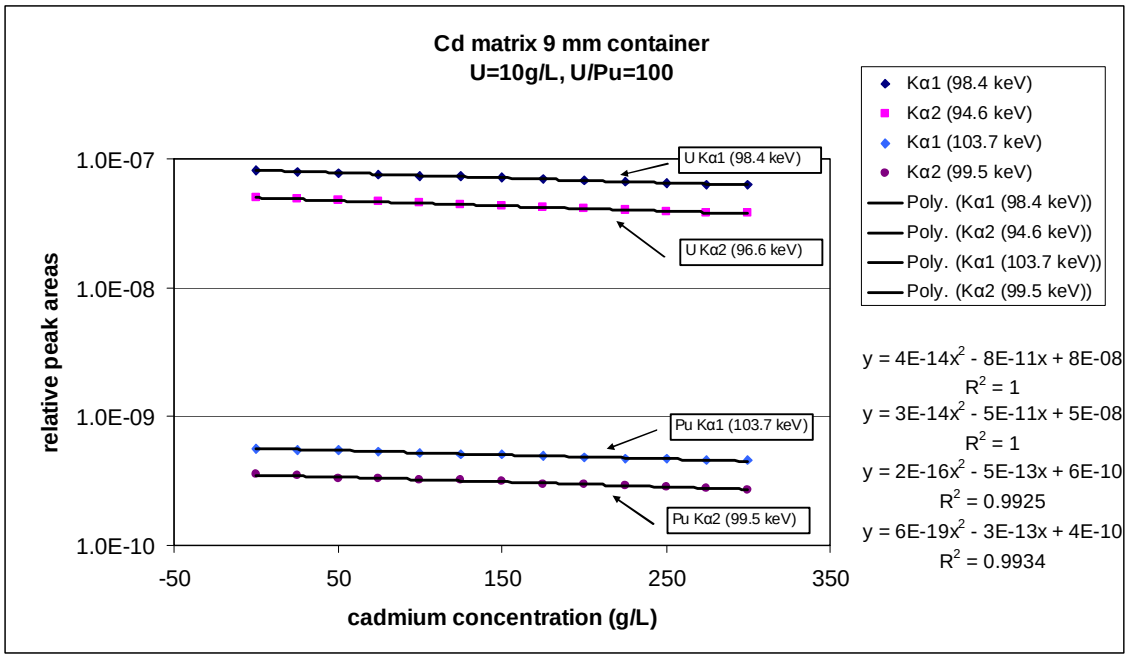


**Figure 8b.** Relative U and peak areas for cadmium samples in a 9 mm container

For the set of bismuth samples, cadmium samples and the set of U+Pu only samples, the *detectability* of the U-Kα and Pu-Kα x-rays, as defined in the previous section, significantly reduces with increasing metal concentration. The smallest reduction is predicted for the cadmium samples, a somewhat larger reduction applies to the U+Pu only samples and the largest reduction is seen in the bismuth samples. However, owing to the similarity in energy of all Kα x-rays concerned, most of that reduction is common to the Kα x-rays of both elements. Hence, the U/Pu peak area ratio is far less affected by changes in metal concentration than are the ratio components themselves. The effect of metal concentration on the U/Pu peak area ratio for all three sample-sets is shown in Fig. 9. Note that the samples in Fig. 9 have different acid molarities, so that their U/Pu value at x = 0 g/L is not identical.

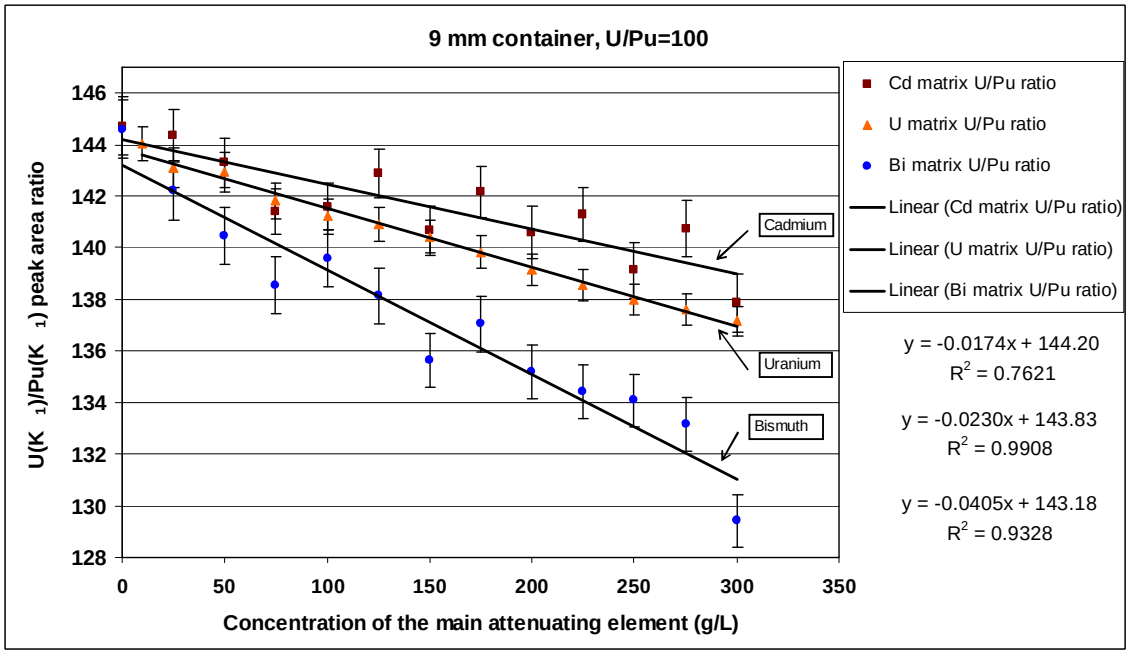


**Figure 9.** Showing the influence of the sample's metal concentration on U/Pu peak area ratios

If we anticipate the metal concentration of all three sample-sets to range from 0—300 g/L and would therefore perform an instrument point-calibration using representative calibration reference material with a metal concentration of ≈150 g/L, then the computed correction polynomials for a given sample-set would only need to be applied to measurement samples with a metal concentration that differs from that of the reference material by, at most, 150 gU/L. Using the case of a 300 g/L sample at the upper-end of the range, the correction factor for the observed U/Pu peak area ratio would then be ≈1.019 for the cadmium matrix, ≈1.025 for the uranium matrix and ≈1.046 for samples with a bismuth matrix. Similarly, the correction factor for cadmium, uranium or bismuth matrices with a metal concentration close to 0 g/L is ≈1.018, ≈1.024 and ≈1.042 respectively. The potential bias in the final measurement results for samples at the extreme end of the concentration range is expected to be ≈0.3%, with both the correction and its potential bias vanishing the closer the characteristics of a measurement sample are to the characteristics of the 150 g/L calibration reference material. Although the 9 mm diameter container is the most likely candidate for measurements on pyro-reprocessing samples, the correction factors are expected to be larger when sample containers with a 14 mm diameter are used. Detailed calculations for the latter geometry have not yet been performed.

The fact that bismuth matrices require the largest corrections is not surprising. Across the entire x-ray energy spectrum from 21 keV to infinity, the x-ray attenuation per unit mass is significantly larger for uranium than for bismuth, except for the narrow energy range between 90.536—115.608 keV, i.e. the energy range between the binding energy of bismuth K-electrons and those of uranium. The main fluorescent Kx-ray emissions from uranium and plutonium, however, have energies that fall exactly within that narrow range and are therefore more affected by changes in bismuth concentration than by changes in uranium concentration. Some relevant details can be found in Table 2.

**Table 2.** Element and energy dependent x-ray mass attenuation coefficients (in $cm^2/g$), including coherent scattering. The data has been taken from XCOM [5].

| x-ray | x-ray energy (keV) | cadmium | uranium | bismuth |
|---|---|---|---|---|
| U Kα2 | 94.653 | 1.760 | 2.236 | 6.592 |
| U Kα1 | 98.434 | 1.588 | 2.031 | 5.971 |
| Pu Kα2 | 99.526 | 1.543 | 1.979 | 5.807 |
| Pu Kα1 | 103.735 | 1.385 | 1.789 | 5.229 |
| molar mass (amu) | | 112.41 | 238.05 | 208.98 |

## 5. Concluding remarks

The spectrum of the x-ray source of the HKED instrument has been determined from appropriate experimental data, combining model calculations with de-convolution. That source-term has been implemented in all models that are now being studied. It has been demonstrated that the traditional multi-point calibration effort can be reduced to a single point-calibration. The accuracy and potential for bias of the modelling approach appears to be no worse than with using a traditional multi-point calibration. Modelling can provide retrospective corrections for exceptionally exotic samples that were never foreseen during the planning stages of a traditional calibration. Modelling can predict corrections for cases in which it is practically impossible to produce appropriate reference material, e.g. highly active waste samples. In that case one could consider using calibration material of a different, but manageable, composition and use model-predicted corrections that refer to the calibration point provided by the alternative reference material. In general, the modelling approach appears flexible enough to be applied to situations that are presently not foreseen. Now that we have gained better understanding of modelling, it is time to validate model predictions against the traditional method of calibration. From the discussions presented here it is evident that bismuth matrices represent the biggest challenge and it is our plan to validate model predictions against measurements on carefully certified bismuth solutions containing U and Pu. This validation is considered a key-objective, and a positive outcome will provide us with enough confidence to apply the single-point calibration approach to all our future measurement efforts. At that stage it will be appropriate to enhance our existing HKED measurement software to incorporate model-based correction factors. Regarding the measurements of pyro-reprocessing samples, we are in the fortunate situation that both the instrument and the modelling are available before the first active samples will arrive.

## Acknowledgements

The authors acknowledge our former colleagues Herbert Ottmar and Heinrich Eberle for developing the Hybrid K-edge / XRF concept and for converting that concept into an instrument that has since become the main workhorse for measuring nuclear material accountancy samples at several spent fuel reprocessing plants around the world; without the Heinrich/Herbert partnership, the HKED instruments would not have existed. Our colleagues now fully deserve their retirement, and we hope that, even without their valuable guidance, we can develop their legacy further to meet present and future measurement challenges.
A special thank you goes to Andrey Berlizov for his valuable comments and for sharing his knowledge of physics and numerical modelling with us.

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# Grey sets theory based approach for assessing the vulnerability of safeguarding a geological repository

**Catrinel Turcanu[1], Lusine Mkrtchyan[2], Klaas van der Meer[1]**

[1]Belgian Nuclear Research Centre SCK•CEN, Boeretang 200, BE 2400 Mol Belgium
[2]Paul Scherrer Institute, 5232 Villigen, Switzerland

E-mail:cturcanu@sckcen.be; kvdmeer@sckcen.be

***Abstract***

*Safeguards approaches for geological repositories are currently under development in the IAEA ASTOR working group and a number of institutions worldwide.*
*In this paper we propose a new method for the assessment of the vulnerability of safeguarding a geological repository. This method draws on recent vulnerability analysis methods proposed in the literature and on multi-criteria analysis theory. In addition, uncertainty in the estimation of the different vulnerability criteria is tackled based on grey theory, which allows expressing uncertainty as intervals of values (grey numbers) and offers ways to adapt known aggregation methods. Finally, we discuss the robustness of various safeguards strategies with respect to five acquisition paths. As an exemplary case study we discuss the operational stage of a geological repository.*

**Keywords:** vulnerability assessment, geological repository, safeguards, uncertainty

## 1. Introduction

Vulnerability analysis originated from the field of natural disasters management, but has since found widespread application in various fields, including security of critical infrastructures. This makes it a promising theoretical paradigm for tackling proliferation resistance issues.

Closely related to the concept of vulnerability is that of resilience [1], which describes the capacity of a system to adapt to changing conditions without catastrophic loss of form or intended function Park et al. [1] point out that resilience demands continuous and adaptive management, awareness of system's incompleteness and new approaches to design (e.g. fail-safe systems). Einarsson and Rausand [2] differentiate between – possibly overlapping - external vulnerability factors and internal vulnerability factors and propose a general taxonomy of risk factors (threats). They also suggest a scenario-based approach, where a scenario is defined as a "sequence of potential events, where the events may be separated in time and space, and where barriers to prevent the sequence are a part of the scenario".

In the framework of the ASTOR task, a joint Support Programme task to the IAEA has the purpose to discuss and develop the safeguards approach for geological repositories, whereby the Finnish design of a geological repository is taken as a reference design. In previous publications [3], [4] we have explored the use of vulnerability analysis in developing a safeguards approach for a geological repository, by proposing vulnerability assessment criteria and an aggregation procedure for obtaining global vulnerability scores. The five diversion paths considered for an open geological repository are described in detail in [3].

In this paper we propose a way to incorporate uncertainties in this analysis by means of grey numbers theory and we discuss some aspects pertaining to robustness analysis. The diversion paths considered are treated as scenarios, whereas the safeguards measures constitute possible decision alternatives that are to be evaluated against the vulnerability assessment criteria.

## 2. Vulnerability assessment

### 2.1 The case without uncertainty

*2.1.1 General framework*

The concept of vulnerability is multi-dimensional, most often the indicators used being qualitative.The studies in the literature show that most attempts at quantifying vulnerability assessment draw on multi-criteria analysis methodology, since it offers a structured framework to deal with multi-factorial problems. This often implies the construction of an overall vulnerability index, e.g. by means of a simple additive aggregation of vulnerability criteria scores weighted according to their perceived importance [5], ]6], and/or the ranking of different scenarios depending on their impact [2].

Drawing on multi-criteria analysis theory, vulnerability of critical infrastructures can be evaluated by means of vulnerability criteria. These can be derived based on INFCIRC/153, stating that the technical purpose of safeguards is: "…*the timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or of other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by the risk of early detection*." In our previous work [3],[4] we have proposed the following vulnerability criteria:

- $C_1$*: Detection robustness*; this combines the concept of detection probability with that of robustness to falsification. This criterion represents the (subjective) probability that a diversion will be detected by the safeguards system, even in case of a falsification.
- $C_2$*: False alarm probability;* this captures the probability that a safeguards measure will trigger an alarm without a real diversion taking place
- $C_3$*: Delay time;* the delay time is the time between occurrence of a diversion and its discovery.

The indicators used to assess the criteria $C_i$ can take values in a certain range $R(C_i)$ which may be expressed in days, probability values, etc. Some criteria may have to be maximized (e.g. detection robustness), others have to be minimized (e.g. false alarm probability and delay time).

For each diversion scenario, the overall vulnerability V of the repository in the context of a safeguards strategy *S* and a diversion scenario *d* can then be calculated as [4]:

$$V(S,d) = 100 - \sum_{i=1}^{3} w_i \cdot f_i(C_i(S,d)),$$

$$\text{with } \sum_{i=1}^{3} w_i = 1 \quad \text{and} \quad f_i : R(C_i) \rightarrow [0, 100], \text{ for } 1 \leq i \leq 3.$$

The quantities in the formulae above will be explained in the following.

The functions $f_i$ are value functions associated to criteria $C_i$, and express the degree of fulfilment of the specific safeguards objective by the particular value $C_i(S,d)$ attained for the safeguards strategy *S* and diversion scenario *d*. These functions are in fact a representation of the different dimensions of the proliferation resistance of the system (the opposite of vulnerability). Figures 1a-1c illustrate the proliferation resistance value functions we propose [4], corresponding to the three criteria.

The weights $w_i$ are related to the importance of the different vulnerability criteria. Multi-criteria analysis theory provides various ways to elicit weights. One of these methods is the Analytical Hierarchical Process (AHP) by Saaty [7], which derives a set of numerical weights compatible with the subjective pairwise comparisons made by an expert. For instance, detection of a diversion can be considered as the most important criterion, since without the detection of a diversion the main safeguards goal is not achieved. The delay time can also be considered as much more important than the false alarm probability. The delay time is directly connected to "timely detection", whereas a higher false alarm probability will cause re-verification activities and has therefore mainly budgetary implications, assuming re-verification can be performed appropriately. For such preferential judgments, AHP yields a weight of 0.714 for the detection probability, 0.223 for the delay time and 0.063 for the false alarm probability. Other methods could also be employed to derive weights. For instance the SWING weighting process starts with determining the criterion for which a change from the lowest possible to

the highest achievable value is of greatest importance. This criterion would then be assigned a (temporary) weight of 100, while the weights for the remaining criteria would be assigned relative to the former criterion, in a similar way; this step would need to be followed by a normalisation of weights' values in order to ensure that they add up to 1.

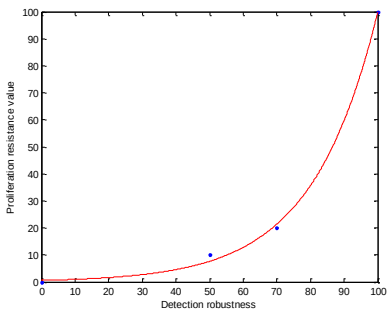


Fig.1 a)

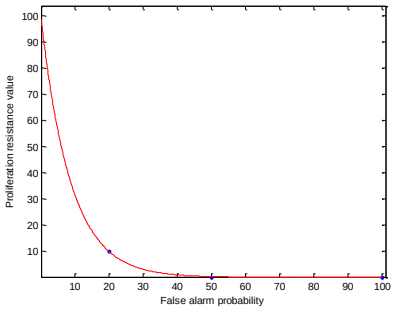


Fig. 1 b)

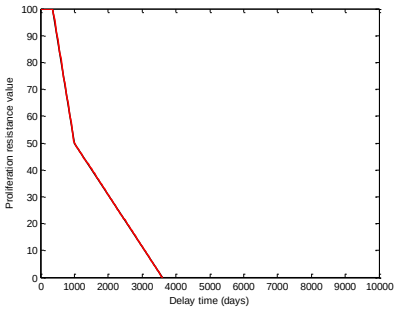


Fig. 1 c)

Fig. 1a) Proliferation resistance value function for detection robustness; exponential fit approximating $f_1(0)=0$, $f_1(50)=10$, $f_1(70)=20$ and $f_1(100)=100$;

1b) Proliferation resistance value function for false alarm probability; exponential fit approximating $f_2(0)=100$, $f_2(10)=20$, $f_2(0.001)=50$, $f_2(100)=0$;

1c) Delay time between diversion and detection, linear interpolation; $f_3(x)=100$ for $x \leq 365$, $f_3(1000)=50$, $f_3(x)=0$ for $x \geq 1000$.

The scale used to evaluate vulnerability is thus 0-100, with a score 0 corresponding to a system that is resistant (completely non vulnerable) and a score of 100 corresponding to a system that is completely vulnerable or inoperable. It can also be seen that minimising vulnerability is equivalent to maximising proliferation resistance. In order to simplify the analysis, we shall focus in the following on the assessment of proliferation resistance, keeping in mind its relation with vulnerability.

*2.1.2 Evaluation of combinations of individual safeguards measures*

Based on a draft safeguards approach for a geological repository [8], a set of safeguards measures was defined for use in the analysis. The set of safeguards measures consists of:

- Nuclear material accountancy (NMA)
- Containment and Surveillance (C/S), application of seals to spent fuel canisters (SE)
- C/S, camera surveillance on movements of spent fuel canisters (CA)
- Monitoring, based on e.g. radiation monitoring, canister identification, directional movement monitoring (MON)
- Design Information Verification (DIV), based on:
  - In situ verification (DIVV)
  - Satellite Imagery (DIVSI)
  - Geophysical Monitoring (DIVGM)

Most combinations of the above-mentioned safeguards measures can be considered as feasible safeguards strategies. NMA was included in all combinations and the in-situ DIV was included in all DIV activities.

In a previous work [4], the evaluation of the different combinations of safeguards measures with respect to the vulnerability criteria ($C_i(S, d)$) has been done by subjective expert assessment. In this paper we propose that the expert evaluates only the individual measures with respect to each vulnerability criterion (detection robustness, false alarm probability and delay time), followed by an automated evaluation of the respective values for combinations of safeguards measures. This proceeds as follows.

In the case of detection robustness, $C_1$, the detection robustness of a combination $S=\{S_1, S_2,.., S_m\}$ of individual measures in the context of a given scenario d is:

$$C_1(S, d) = 1- (1-C_1(S_1 , d)) \cdot (1-C_1(S_2 , d) \cdot \ldots \cdot (1-C_1(S_m, d))$$

The motivation is that from the point of view of detection robustness, a combination of safeguards measures would fail (the diversion remains undetected) only if all individual measures fail simultaneously.

For the false alarm probability, we assumed that each individual measure can generate independently a false alarm. However, we consider that a false alarm in the usual meaning of the concept would occur only if all individual measures are generating this false alarm simultaneously (in other cases they would only lead to verification). Thus, for any scenario d:

$$C_2(S,d) = C_2(S_1, d) \cdot C_2(S_2, d) \cdot \ldots \cdot C_2(S_m, d).$$

In the case of the delay time, $C_3$, we considered that as soon as one of the individual measures would raise an alarm, the diversion would be detected. Therefore:

$$C_3(S,d) = \min \{ C_3(S_1, d), C_3(S_2, d), \ldots, C_3(S_m, d) \}.$$

## 2.2. The case with uncertainty

In order to account for the inherent uncertainty in the evaluation of the different individual or combined safeguards measures with respect to the vulnerability criteria, we propose the use of intervals specifying the lowest and respectively the highest achievable value. We also assume that there is no complete knowledge about probability distributions on these intervals of values.

In the literature, there are several widely used methods dealing with uncertainty, the most important being probability theory, fuzzy set theory, rough set theory and grey set theory. While the first two dominate the relevant literature, one has to notice that these methods address different types of uncertainty. For example, the choice between the use of fuzzy set theory and grey set theory depends if the problem at hand has cognitive uncertainties or it is characterised by poor information. The main difference between the two theories is that grey set theory focuses on the problems where boundaries are clear, but the exact values are not known (whereas fuzzy sets theory addresses problems where boundaries are unclear). With respect to probabilistic analysis, grey set theory is given a priority if the sample in the problem is small, as for probabilistic and statistical analysis the availability of large samples is assumed [9].

Thus, a suitable theoretical framework for the problem described here the theory of grey sets [10]. In grey systems, the information is classified into three categories: white (completely certain), grey (insufficient information) and black (totally unknown). A grey number $G$ is a number with clear lower and upper boundaries, but whose exact position is not known; in other words it is a set of discrete or continuous values lying in between known boundaries. When the grey number is reduced to an interval $G=[G^-, G^+]$, defined by its upper and lower bound, the arithmetic applied is the usual arithmetic of intervals numbers [11]:

$[G_1^-, G_1^+] + [G_2^-, G_2^+] = [G_1^- + G_2^-, G_1^+ + G_2^+]$ ;

$[G_1^-, G_1^+] - [G_2^-, G_2^+] = [G_1^- - G_2^+, G_1^+ - G_2^-]$ ;

$[G_1^-,G_1^+] \times [G_2^-,G_2^+] = [\min \{G_1^- \cdot G_2^-, G_1^- \cdot G_2^+, G_1^+ \cdot G_2^-, G_1^+ \cdot G_2^+\}, \max \{G_1^- \cdot G_2^-, G_1^- \cdot G_2^+, G_1^+ \cdot G_2^-, G_1^+ \cdot G_2^+\}]$

$[G_1^-, G_1^+] \div [G_2^-, G_2^+] = [G_1^-, G_1^+] \times [1/G_2^+, 1/G_2^-]$, provided that $0 \notin [G_2^-, G_2^+]$ ;

$k \cdot [G_1^-, G_1^+] = [k \cdot G_1^-, k \cdot G_1^+]$, for any $k>0$.

By making use of these operations, we can extend the vulnerability analysis described in section 2.1 to the case with uncertainty. In this way, we can evaluate all individual and combined measures with respect to the three vulnerability criteria and subsequently we can calculate the proliferation resistance values scores and the aggregated proliferation resistance value in a similar way as for the case without uncertainty, but using intervals this time.

Table 1 illustrates the calculation of the vulnerability values for the cases with and without uncertainty for one selected scenario (scenario 1, noted d1). Since the false alarm probability is a criterion that has to be minimised, the proliferation resistance value score corresponding to the lower bound of the false alarm probability is of course higher than the value score corresponding to the upper bound. The same is valid for the delay time.

## 2.3 Deriving robust recommendations

Similar calculations as those presented in Table 1 have to be performed for each of the diversion scenarios that have to be included in the analysis. We therefore need not only to assess the different measures in the context of a single diversion scenario, but to evaluate their efficiency across all five scenarios. The next step question is to evaluate in structured way which measures perform efficiently in all scenarios.

For this purpose we make again use of multi-criteria analysis theory, namely robustness analysis. In the optimisation and decision aid domains, the notion of robustness may have different interpretations [12]. In strategic decisions involving sequential decision-making Rosenhead et al [13] defined robustness as a measure of flexibility, expressing the potential of decision taken at a given moment to allow for achieving near-optimal states in the future, in conditions of uncertainty. Kouvelis and Yu [14] defined several measures of robustness.

For instance, an absolute robust solution would be one that satisfies:

$$X_r^{abs} = \arg \min_S \max_d V(S, d),$$

where S denotes safeguards strategies, d denotes diversion paths and V(S, d) denotes the vulnerability of the repository if strategy S is applied in the context of the diversion path d.

This measure is very conservative, so the following definition, describing a deviation robust solution, might be preferred:

$$X_r^{dev} = \arg \min_S \max_d (V(S,d) - V(S_d^*,d)),$$

where $S_d^*$ is the optimal safeguards strategy for diversion scenario d.

A relation similar to the one above can be written for proliferation resistance, given its relation with vulnerability. In the case when criteria *Ci* are evaluated by means of intervals (the case with uncertainty), for each scenario d, the ideal in terms of proliferation resistance can be either represented by the final (but not reachable) goal of (100,100) as proliferation value scores (lower bound= upper bound = total proliferation resistance) or as the best achievable lower bound and upper bound, respectively (e.g. [max(DRV_lower), max(DRV_higher)] with the notations from Table 1.

The deviation from the ideal in terms of proliferation resistance can be calculated, for each scenario, using the weighted Minkowski metric.

For a safeguards strategy S that is evaluated as $[C_i(S, d)^-, C_i(S,d)^+]$ with respect to criteria $C_i$, $1 \le i \le 3$, in the context of scenario *d*, the distance to the ideal S* corresponding this scenario can be written as:

$$D(S, S^*) = \left(1/2 \cdot \Sigma_i w_i \cdot [\, |f_i(C_i(S, d)^-) - f_i(C_i(S^*)^-)|^p + |\, f_i(C_i(S, d)^+) - f_i(C_i(S^*)^+)|^p \,]\right)^{1/p}, \text{ where}$$

$$f_i(C_i(S^*)^-) = \max_S f_i(C_i(S, d)^-) \text{ and } f_i(C_i(S^*)^+) = \max_S f_i(C_i(S, d)^+)$$

In our example we have used a value p=2. Note that in the formula above we use the proliferation value scores $f_i(C_i(\cdot\,,\cdot))$ instead of the $C_i(\cdot\,,\cdot)$, in order to work with values belonging to the same range (here between 0 and 100). The weights $w_i$ used are the same as before.

Further on, a ranking of safeguards measures can be obtained based on the maximal distance to the ideal, across all scenarios considered (deviation robust approach). Figure 2 sums up the steps we have employed for vulnerability assessment, which is applicable for both cases with and without uncertainty.

**Table 1 Proliferation resistance assessment for scenario d1**

(DR=detection robustness, FA= false alarm probability, Ti= delay time, DRV=values score for DR, FAV=value score for FA, TiV=value score for Ti)
Grey background= value given by expert; white background= value calculated

| NMA | SE | CA | MON | DIVV | DIVSI | DIVGM | DR-Lower | DR-Higher | FA-Lower | FA-Higher | Ti-Lower | Ti-Higher | DRV-Lower | DRV-Higher | FAV-Lower | FAV-Higher | TiV-Lower | TiV-Higher |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0.00 | 0.05 | 0.00 | 0.10 | 1800 | 5400 | 0.58 | 0.75 | 100.00 | 31.66 | 34.60 | 0.00 |
| 0 | 1 | 0 | 0 | 0 | 0 | 0 | 0.30 | 0.70 | 0.00 | 0.02 | 0 | 360 | 2.70 | 20.74 | 100.00 | 79.45 | 100.00 | 100.00 |
| 0 | 0 | 1 | 0 | 0 | 0 | 0 | 0.30 | 0.70 | 0.03 | 0.07 | 15 | 45 | 2.70 | 20.74 | 70.82 | 44.71 | 100.00 | 100.00 |
| 0 | 0 | 0 | 1 | 0 | 0 | 0 | 0.20 | 0.60 | 0.04 | 0.06 | 15 | 45 | 1.62 | 12.46 | 63.13 | 50.16 | 100.00 | 100.00 |
| 0 | 0 | 0 | 0 | 1 | 0 | 0 | 0.00 | 0.05 | 0.00 | 0.00 | 1800 | 5400 | 0.58 | 0.75 | 100.00 | 100.00 | 34.60 | 0.00 |
| 0 | 0 | 0 | 0 | 0 | 1 | 0 | 0.00 | 0.10 | 0.00 | 0.00 | 1800 | 5400 | 0.58 | 0.97 | 100.00 | 100.00 | 34.60 | 0.00 |
| 0 | 0 | 0 | 0 | 0 | 0 | 1 | 0.00 | 0.00 | 0.00 | 0.00 | 1800 | 5400 | 0.58 | 0.58 | 100.00 | 100.00 | 34.60 | 0.00 |
| 1 | 1 | 0 | 0 | 0 | 0 | 0 | 0.30 | 0.72 | 0.00 | 0.00 | 0 | 360 | 2.70 | 22.39 | 100.00 | 97.73 | 100.00 | 100.00 |
| 1 | 0 | 1 | 0 | 0 | 0 | 0 | 0.30 | 0.72 | 0.00 | 0.01 | 15 | 45 | 2.70 | 22.39 | 100.00 | 92.27 | 100.00 | 100.00 |
| 1 | 1 | 1 | 0 | 0 | 0 | 0 | 0.51 | 0.91 | 0.00 | 0.00 | 0 | 45 | 7.87 | 61.94 | 100.00 | 99.84 | 100.00 | 100.00 |
| 1 | 0 | 0 | 1 | 0 | 0 | 0 | 0.20 | 0.62 | 0.00 | 0.01 | 15 | 45 | 1.62 | 13.79 | 100.00 | 93.33 | 100.00 | 100.00 |
| 1 | 1 | 0 | 1 | 0 | 0 | 0 | 0.44 | 0.89 | 0.00 | 0.00 | 0 | 45 | 5.51 | 53.56 | 100.00 | 99.86 | 100.00 | 100.00 |
| 1 | 0 | 1 | 1 | 0 | 0 | 0 | 0.44 | 0.89 | 0.00 | 0.00 | 15 | 45 | 5.51 | 53.56 | 100.00 | 99.52 | 100.00 | 100.00 |
| 1 | 1 | 1 | 1 | 0 | 0 | 0 | 0.61 | 0.97 | 0.00 | 0.00 | 0 | 45 | 12.97 | 80.46 | 100.00 | 99.99 | 100.00 | 100.00 |
| 1 | 0 | 0 | 0 | 1 | 0 | 0 | 0.00 | 0.10 | 0.00 | 0.00 | 1800 | 5400 | 0.58 | 0.96 | 100.00 | 100.00 | 34.60 | 0.00 |
| 1 | 1 | 0 | 0 | 1 | 0 | 0 | 0.30 | 0.73 | 0.00 | 0.00 | 0 | 360 | 2.70 | 24.08 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 0 | 1 | 0 | 1 | 0 | 0 | 0.30 | 0.73 | 0.00 | 0.00 | 15 | 45 | 2.70 | 24.08 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 1 | 1 | 0 | 1 | 0 | 0 | 0.51 | 0.92 | 0.00 | 0.00 | 0 | 45 | 7.87 | 63.30 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 0 | 0 | 1 | 1 | 0 | 0 | 0.20 | 0.64 | 0.00 | 0.00 | 15 | 45 | 1.62 | 15.20 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 1 | 0 | 1 | 1 | 0 | 0 | 0.44 | 0.89 | 0.00 | 0.00 | 0 | 45 | 5.51 | 55.14 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 0 | 1 | 1 | 1 | 0 | 0 | 0.44 | 0.89 | 0.00 | 0.00 | 15 | 45 | 5.51 | 55.14 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 1 | 1 | 1 | 1 | 0 | 0 | 0.61 | 0.97 | 0.00 | 0.00 | 0 | 45 | 12.97 | 81.16 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 0 | 0 | 0 | 1 | 1 | 0 | 0.00 | 0.19 | 0.00 | 0.00 | 1800 | 5400 | 0.58 | 1.52 | 100.00 | 100.00 | 34.60 | 0.00 |
| 1 | 1 | 0 | 0 | 1 | 1 | 0 | 0.30 | 0.76 | 0.00 | 0.00 | 0 | 360 | 2.70 | 27.64 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 0 | 1 | 0 | 1 | 1 | 0 | 0.30 | 0.76 | 0.00 | 0.00 | 15 | 45 | 2.70 | 27.64 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 1 | 1 | 0 | 1 | 1 | 0 | 0.51 | 0.93 | 0.00 | 0.00 | 0 | 45 | 7.87 | 65.98 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 0 | 0 | 1 | 1 | 1 | 0 | 0.20 | 0.68 | 0.00 | 0.00 | 15 | 45 | 1.62 | 18.27 | 100.00 | 100.00 | 100.00 | 100.00 |

| NMA | SE | CA | MON | DIVV | DIVSI | DIVGM | DR-Lower | DR-Higher | FA-Lower | FA-Higher | Ti-Lower | Ti-Higher | DRV-Lower | DRV-Higher | FAV-Lower | FAV-Higher | TiV-Lower | TiV-Higher |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 1 | 0 | 1 | 1 | 1 | 0 | 0.44 | 0.90 | 0.00 | 0.00 | 0 | 45 | 5.51 | 58.27 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 0 | 1 | 1 | 1 | 1 | 0 | 0.44 | 0.90 | 0.00 | 0.00 | 15 | 45 | 5.51 | 58.27 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 1 | 1 | 1 | 1 | 1 | 0 | 0.61 | 0.97 | 0.00 | 0.00 | 0 | 45 | 12.97 | 82.52 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 0 | 0 | 0 | 1 | 0 | 1 | 0.00 | 0.10 | 0.00 | 0.00 | 1800 | 5400 | 0.58 | 0.96 | 100.00 | 100.00 | 34.60 | 0.00 |
| 1 | 1 | 0 | 0 | 1 | 0 | 1 | 0.30 | 0.73 | 0.00 | 0.00 | 0 | 360 | 2.70 | 24.08 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 0 | 1 | 0 | 1 | 0 | 1 | 0.30 | 0.73 | 0.00 | 0.00 | 15 | 45 | 2.70 | 24.08 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 1 | 1 | 0 | 1 | 0 | 1 | 0.51 | 0.92 | 0.00 | 0.00 | 0 | 45 | 7.87 | 63.30 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 0 | 0 | 1 | 1 | 0 | 1 | 0.20 | 0.64 | 0.00 | 0.00 | 15 | 45 | 1.62 | 15.20 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 1 | 0 | 1 | 1 | 0 | 1 | 0.44 | 0.89 | 0.00 | 0.00 | 0 | 45 | 5.51 | 55.14 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 0 | 1 | 1 | 1 | 0 | 1 | 0.44 | 0.89 | 0.00 | 0.00 | 15 | 45 | 5.51 | 55.14 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 1 | 1 | 1 | 1 | 0 | 1 | 0.61 | 0.97 | 0.00 | 0.00 | 0 | 45 | 12.97 | 81.16 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 0 | 0 | 0 | 1 | 1 | 1 | 0.00 | 0.19 | 0.00 | 0.00 | 1800 | 5400 | 0.58 | 1.52 | 100.00 | 100.00 | 34.60 | 0.00 |
| 1 | 1 | 0 | 0 | 1 | 1 | 1 | 0.30 | 0.76 | 0.00 | 0.00 | 0 | 360 | 2.70 | 27.64 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 0 | 1 | 0 | 1 | 1 | 1 | 0.30 | 0.76 | 0.00 | 0.00 | 15 | 45 | 2.70 | 27.64 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 1 | 1 | 0 | 1 | 1 | 1 | 0.51 | 0.93 | 0.00 | 0.00 | 0 | 45 | 7.87 | 65.98 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 0 | 0 | 1 | 1 | 1 | 1 | 0.20 | 0.68 | 0.00 | 0.00 | 15 | 45 | 1.62 | 18.27 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 1 | 0 | 1 | 1 | 1 | 1 | 0.44 | 0.90 | 0.00 | 0.00 | 0 | 45 | 5.51 | 58.27 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 0 | 1 | 1 | 1 | 1 | 1 | 0.44 | 0.90 | 0.00 | 0.00 | 15 | 45 | 5.51 | 58.27 | 100.00 | 100.00 | 100.00 | 100.00 |
| 1 | 1 | 1 | 1 | 1 | 1 | 1 | 0.61 | 0.97 | 0.00 | 0.00 | 0 | 45 | 12.97 | 82.52 | 100.00 | 100.00 | 100.00 | 100.00 |

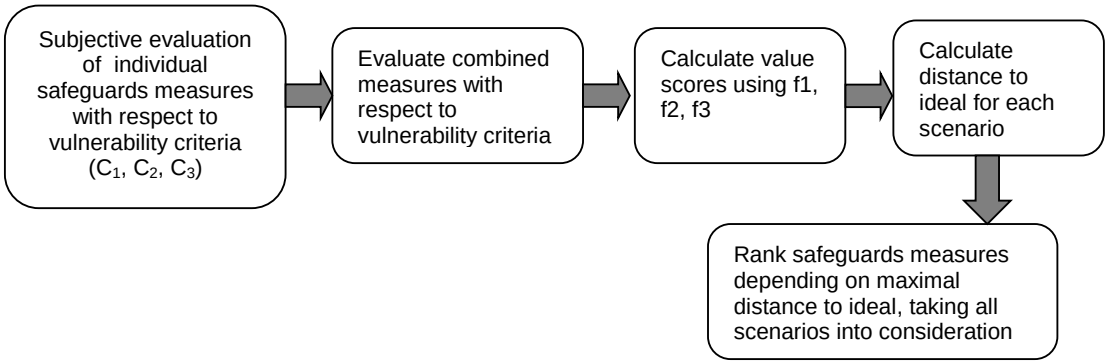


**Fig. 2 Steps of vulnerability assessment**

Table 2 presents the aggregated proliferation resistance value scores corresponding to each criterion, the deviation from the respective ideal in terms of proliferation resistance, and the maximal deviation.

Looking to these results from a safeguards point of view it is not surprising that the full combination of safeguards measures leads to the best value of proliferation resistance and thus to the lowest value of the maximum deviation from best achievable scores across all scenarios.

However, what can be useful for the safeguards inspectorates is the possibility to use the value for the maximum deviation to optimise the safeguards approach with respect to inspection costs, without losing significantly in safeguards effectiveness.

From this example we see clearly that DIV by satellite imagery does not add significantly to the overall effectiveness. The elimination of the second “worst” safeguards measure, C/S by seals, results in an 8 times higher value of the maximum deviation resulting from eliminating satellite imagery instead. Elimination of other safeguards measures results in comparable and even higher increases. From the point of view of safeguards effectiveness we can deduce based on this example that it is less obvious to eliminate a measure such as application of seals than satellite imagery.

Another interesting insight comes from the comparison of safeguards strategies containing the application of seals versus those that include the application of cameras. The maximum deviation scores of the strategies compared for this purpose are highlighted in bold font in the last column in table 2. The analysis of these values shows a large difference in the maximum deviation score when neither of the considered safeguards measures is combined with the application of monitoring, while the difference is much less or not present when monitoring is applied. Since the application of seals results in the higher maximum deviation scores than the application of cameras – particularly in the context of the second and third diversion paths - it can therefore be considered as less safeguards effective overall. From this we can conclude that in the case when monitoring cannot be implemented, for one or another reason, the application of cameras should be preferred to the application of seals from the point of view of safeguards effectiveness.

The method provides also the possibility to look to the detailed results for the various diversion scenarios. This allows the assessment of the effectiveness of individual safeguards measures on the overall safeguards effectiveness.

**Table 2 Aggregated proliferation resistance value scores and deviation from ideal in each scenario**
(di-L= lower bound for aggregated proliferation resistance score, di-H= upper bound for aggregated proliferation resistance score, di-D2Opt=deviation from ideal in the case of scenario di, MAX=maximal deviation considering all scenarios)

| NMA | SE | CA | MON | DIVV | DIVSI | DIVGM | d1-L | d1-H | d1-D2Opt | d2-L | d2-H | d2-D2Opt | d3-L | d3-H | d3-D2Opt | d4-L | d4-H | d4-D2Opt | d5-L | d5-H | d5-D2Opt | MAX |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 2.4 | 14.6 | 45.0 | 2.4 | 14.6 | 43.1 | 2.4 | 14.6 | 36.0 | 2.4 | 14.6 | 52.9 | 2.4 | 14.6 | 53.8 | 53.8 |
| 0.0 | 1.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 29.2 | 43.4 | 23.0 | 6.7 | 14.4 | 41.0 | 6.7 | 14.4 | 33.9 | 6.7 | 14.4 | 50.8 | 6.7 | 14.4 | 51.7 | 51.7 |
| 0.0 | 0.0 | 1.0 | 0.0 | 0.0 | 0.0 | 0.0 | 27.0 | 41.6 | 24.3 | 27.0 | 41.6 | 22.6 | 26.3 | 31.1 | 20.6 | 6.7 | 14.4 | 50.8 | 6.7 | 14.4 | 51.7 | 51.7 |
| 0.0 | 0.0 | 0.0 | 1.0 | 0.0 | 0.0 | 0.0 | 26.6 | 35.2 | 27.3 | 34.4 | 50.9 | 16.8 | 26.6 | 50.9 | 10.4 | 6.7 | 14.4 | 50.8 | 6.7 | 14.4 | 51.7 | 51.7 |
| 0.0 | 0.0 | 0.0 | 0.0 | 1.0 | 0.0 | 0.0 | 6.7 | 14.6 | 42.8 | 6.7 | 14.6 | 40.9 | 6.7 | 14.6 | 33.9 | 58.6 | 97.0 | 0.9 | 28.7 | 43.4 | 32.7 | 42.8 |
| 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 1.0 | 0.0 | 6.7 | 14.7 | 42.8 | 6.7 | 14.7 | 40.8 | 6.7 | 14.7 | 33.8 | 6.7 | 14.4 | 50.8 | 14.0 | 30.5 | 45.3 | 50.8 |
| 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 1.0 | 6.7 | 14.4 | 42.9 | 6.7 | 14.4 | 41.0 | 6.7 | 14.4 | 33.9 | 6.7 | 14.4 | 50.8 | 49.8 | 96.3 | 7.4 | 50.8 |
| 1.0 | 1.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 30.4 | 44.6 | 21.8 | 6.7 | 14.6 | 40.9 | 6.7 | 14.6 | 33.9 | 6.7 | 14.6 | 50.7 | 6.7 | 14.6 | 51.6 | **51.6** |
| 1.0 | 0.0 | 1.0 | 0.0 | 0.0 | 0.0 | 0.0 | 30.0 | 44.6 | 22.0 | 30.0 | 44.6 | 20.3 | 28.7 | 32.3 | 19.1 | 6.7 | 14.6 | 50.7 | 6.7 | 14.6 | 51.6 | **51.6** |
| 1.0 | 1.0 | 1.0 | 0.0 | 0.0 | 0.0 | 0.0 | 34.2 | 72.8 | 7.6 | 30.5 | 44.6 | 20.0 | 29.0 | 32.3 | 18.9 | 6.7 | 14.6 | 50.7 | 6.7 | 14.6 | 51.6 | 51.6 |
| 1.0 | 0.0 | 0.0 | 1.0 | 0.0 | 0.0 | 0.0 | 29.3 | 38.4 | 25.1 | 37.1 | 54.6 | 14.4 | 29.3 | 54.6 | 8.0 | 6.7 | 14.6 | 50.7 | 6.7 | 14.6 | 51.6 | 51.6 |
| 1.0 | 1.0 | 0.0 | 1.0 | 0.0 | 0.0 | 0.0 | 32.5 | 66.8 | 10.7 | 37.5 | 54.6 | 14.2 | 29.8 | 54.6 | 7.8 | 6.7 | 14.6 | 50.7 | 6.7 | 14.6 | 51.6 | **51.6** |
| 1.0 | 0.0 | 1.0 | 1.0 | 0.0 | 0.0 | 0.0 | 32.5 | 66.8 | 10.7 | 45.0 | 79.7 | 1.1 | 29.7 | 66.8 | 1.7 | 6.7 | 14.6 | 50.7 | 6.7 | 14.6 | 51.6 | **51.6** |
| 1.0 | 1.0 | 1.0 | 1.0 | 0.0 | 0.0 | 0.0 | 37.9 | 86.0 | 0.7 | 45.0 | 79.7 | 1.1 | 29.8 | 66.8 | 1.7 | 6.7 | 14.6 | 50.7 | 6.7 | 14.6 | 51.6 | 51.6 |
| 1.0 | 0.0 | 0.0 | 0.0 | 1.0 | 0.0 | 0.0 | 6.7 | 14.7 | 42.8 | 6.7 | 14.7 | 40.8 | 6.7 | 14.7 | 33.8 | 60.2 | 97.0 | 0.1 | 30.3 | 44.6 | 31.4 | 42.8 |
| 1.0 | 1.0 | 0.0 | 0.0 | 1.0 | 0.0 | 0.0 | 30.5 | 45.8 | 21.2 | 6.7 | 14.7 | 40.8 | 6.7 | 14.7 | 33.8 | 60.4 | 97.0 | 0.0 | 30.5 | 44.6 | 31.2 | **40.8** |
| 1.0 | 0.0 | 1.0 | 0.0 | 1.0 | 0.0 | 0.0 | 30.5 | 45.8 | 21.2 | 30.5 | 45.8 | 19.5 | 29.0 | 32.9 | 18.6 | 60.4 | 97.0 | 0.0 | 30.5 | 44.6 | 31.2 | **31.2** |
| 1.0 | 1.0 | 1.0 | 0.0 | 1.0 | 0.0 | 0.0 | 34.2 | 73.8 | 7.1 | 30.5 | 45.8 | 19.5 | 29.0 | 32.9 | 18.6 | 60.4 | 97.0 | 0.0 | 30.5 | 44.6 | 31.2 | 31.2 |
| 1.0 | 0.0 | 0.0 | 1.0 | 1.0 | 0.0 | 0.0 | 29.8 | 39.5 | 24.4 | 37.5 | 55.8 | 13.6 | 29.8 | 55.8 | 7.2 | 60.4 | 97.0 | 0.0 | 30.5 | 44.6 | 31.2 | 31.2 |
| 1.0 | 1.0 | 0.0 | 1.0 | 1.0 | 0.0 | 0.0 | 32.5 | 68.0 | 10.1 | 37.5 | 55.8 | 13.6 | 29.8 | 55.8 | 7.2 | 60.4 | 97.0 | 0.0 | 30.5 | 44.6 | 31.2 | **31.2** |
| 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 0.0 | 0.0 | 32.5 | 68.0 | 10.1 | 45.0 | 80.5 | 0.7 | 29.8 | 68.0 | 1.1 | 60.4 | 97.0 | 0.0 | 30.5 | 44.6 | 31.2 | **31.2** |
| 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 0.0 | 0.0 | 37.9 | 86.6 | 0.5 | 45.0 | 80.5 | 0.7 | 29.8 | 68.0 | 1.1 | 60.4 | 97.0 | 0.0 | 30.5 | 44.6 | 31.2 | 31.2 |
| 1.0 | 0.0 | 0.0 | 0.0 | 1.0 | 1.0 | 0.0 | 6.7 | 15.1 | 42.6 | 6.7 | 15.1 | 40.6 | 6.7 | 15.1 | 33.6 | 60.4 | 97.0 | 0.0 | 31.3 | 53.3 | 27.4 | 42.6 |
| 1.0 | 1.0 | 0.0 | 0.0 | 1.0 | 1.0 | 0.0 | 30.5 | 48.3 | 19.9 | 6.7 | 15.1 | 40.6 | 6.7 | 15.1 | 33.6 | 60.4 | 97.0 | 0.0 | 31.4 | 53.3 | 27.4 | **40.6** |
| 1.0 | 0.0 | 1.0 | 0.0 | 1.0 | 1.0 | 0.0 | 30.5 | 48.3 | 19.9 | 30.5 | 48.3 | 18.3 | 29.0 | 34.3 | 18.0 | 60.4 | 97.0 | 0.0 | 31.4 | 53.3 | 27.4 | **27.4** |
| 1.0 | 1.0 | 1.0 | 0.0 | 1.0 | 1.0 | 0.0 | 34.2 | 75.7 | 6.2 | 30.5 | 48.3 | 18.3 | 29.0 | 34.3 | 18.0 | 60.4 | 97.0 | 0.0 | 31.4 | 53.3 | 27.4 | 27.4 |
| 1.0 | 0.0 | 0.0 | 1.0 | 1.0 | 1.0 | 0.0 | 29.8 | 41.6 | 23.3 | 37.5 | 58.5 | 12.3 | 29.8 | 58.5 | 5.9 | 60.4 | 97.0 | 0.0 | 31.4 | 53.3 | 27.4 | 27.4 |
| 1.0 | 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 0.0 | 32.5 | 70.2 | 9.1 | 37.5 | 58.5 | 12.3 | 29.8 | 58.5 | 5.9 | 60.4 | 97.0 | 0.0 | 31.4 | 53.3 | 27.4 | **27.4** |
| 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 1.0 | 0.0 | 32.5 | 70.2 | 9.1 | 45.0 | 81.9 | 0.0 | 29.8 | 70.2 | 0.0 | 60.4 | 97.0 | 0.0 | 31.4 | 53.3 | 27.4 | **27.4** |

| NMA | SE | CA | MON | DIVV | DIVSI | DIVGM | d1-L | d1-H | d1-D2Opt | d2-L | d2-H | d2-D2Opt | d3-L | d3-H | d3-D2Opt | d4-L | d4-H | d4-D2Opt | d5-L | d5-H | d5-D2Opt | MAX |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 0.0 | 37.9 | 87.5 | 0.0 | 45.0 | 81.9 | 0.0 | 29.8 | 70.2 | 0.0 | 60.4 | 97.0 | 0.0 | 31.4 | 53.3 | 27.4 | 27.4 |
| 1.0 | 0.0 | 0.0 | 0.0 | 1.0 | 0.0 | 1.0 | 6.7 | 14.7 | 42.8 | 6.7 | 14.7 | 40.8 | 6.7 | 14.7 | 33.8 | 60.4 | 97.0 | 0.0 | 62.1 | 97.0 | 1.2 | 42.8 |
| 1.0 | 1.0 | 0.0 | 0.0 | 1.0 | 0.0 | 1.0 | 30.5 | 45.8 | 21.2 | 6.7 | 14.7 | 40.8 | 6.7 | 14.7 | 33.8 | 60.4 | 97.0 | 0.0 | 62.1 | 97.0 | 1.2 | **40.8** |
| 1.0 | 0.0 | 1.0 | 0.0 | 1.0 | 0.0 | 1.0 | 30.5 | 45.8 | 21.2 | 30.5 | 45.8 | 19.5 | 29.0 | 32.9 | 18.6 | 60.4 | 97.0 | 0.0 | 62.1 | 97.0 | 1.2 | **21.2** |
| 1.0 | 1.0 | 1.0 | 0.0 | 1.0 | 0.0 | 1.0 | 34.2 | 73.8 | 7.1 | 30.5 | 45.8 | 19.5 | 29.0 | 32.9 | 18.6 | 60.4 | 97.0 | 0.0 | 62.1 | 97.0 | 1.2 | 19.5 |
| 1.0 | 0.0 | 0.0 | 1.0 | 1.0 | 0.0 | 1.0 | 29.8 | 39.5 | 24.4 | 37.5 | 55.8 | 13.6 | 29.8 | 55.8 | 7.2 | 60.4 | 97.0 | 0.0 | 62.1 | 97.0 | 1.2 | 24.4 |
| 1.0 | 1.0 | 0.0 | 1.0 | 1.0 | 0.0 | 1.0 | 32.5 | 68.0 | 10.1 | 37.5 | 55.8 | 13.6 | 29.8 | 55.8 | 7.2 | 60.4 | 97.0 | 0.0 | 62.1 | 97.0 | 1.2 | **13.6** |
| 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 0.0 | 1.0 | 32.5 | 68.0 | 10.1 | 45.0 | 80.5 | 0.7 | 29.8 | 68.0 | 1.1 | 60.4 | 97.0 | 0.0 | 62.1 | 97.0 | 1.2 | **10.1** |
| 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 0.0 | 1.0 | 37.9 | 86.6 | 0.5 | 45.0 | 80.5 | 0.7 | 29.8 | 68.0 | 1.1 | 60.4 | 97.0 | 0.0 | 62.1 | 97.0 | 1.2 | 1.2 |
| 1.0 | 0.0 | 0.0 | 0.0 | 1.0 | 1.0 | 1.0 | 6.7 | 15.1 | 42.6 | 6.7 | 15.1 | 40.6 | 6.7 | 15.1 | 33.6 | 60.4 | 97.0 | 0.0 | 64.6 | 97.0 | 0.0 | 42.6 |
| 1.0 | 1.0 | 0.0 | 0.0 | 1.0 | 1.0 | 1.0 | 30.5 | 48.3 | 19.9 | 6.7 | 15.1 | 40.6 | 6.7 | 15.1 | 33.6 | 60.4 | 97.0 | 0.0 | 64.6 | 97.0 | 0.0 | **40.6** |
| 1.0 | 0.0 | 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 30.5 | 48.3 | 19.9 | 30.5 | 48.3 | 18.3 | 29.0 | 34.3 | 18.0 | 60.4 | 97.0 | 0.0 | 64.6 | 97.0 | 0.0 | **19.9** |
| 1.0 | 1.0 | 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 34.2 | 75.7 | 6.2 | 30.5 | 48.3 | 18.3 | 29.0 | 34.3 | 18.0 | 60.4 | 97.0 | 0.0 | 64.6 | 97.0 | 0.0 | 18.3 |
| 1.0 | 0.0 | 0.0 | 1.0 | 1.0 | 1.0 | 1.0 | 29.8 | 41.6 | 23.3 | 37.5 | 58.5 | 12.3 | 29.8 | 58.5 | 5.9 | 60.4 | 97.0 | 0.0 | 64.6 | 97.0 | 0.0 | 23.3 |
| 1.0 | 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 1.0 | 32.5 | 70.2 | 9.1 | 37.5 | 58.5 | 12.3 | 29.8 | 58.5 | 5.9 | 60.4 | 97.0 | 0.0 | 64.6 | 97.0 | 0.0 | **12.3** |
| 1.0 | 0.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 32.5 | 70.2 | 9.1 | 45.0 | 81.9 | 0.0 | 29.8 | 70.2 | 0.0 | 60.4 | 97.0 | 0.0 | 64.6 | 97.0 | 0.0 | **9.1** |
| 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 37.9 | 87.5 | 0.0 | 45.0 | 81.9 | 0.0 | 29.8 | 70.2 | 0.0 | 60.4 | 97.0 | 0.0 | 64.6 | 97.0 | 0.0 | 0.0 |

## 3. Conclusions

The described methodology requires less elaborated input from the safeguards evaluator than methodologies previously published. A clearer discussion about the underlying assumptions can therefore be held.

The given example shows that the proposed methodology has the potential to evaluate in a relatively easy way the safeguards effectiveness of a set of safeguards measures for various diversion scenarios.

As a future extension of this study we intend to perform a comparative analysis of various theoretical frameworks of uncertainty modelling applied to this problem.

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# An Overview of Research into Arms Control Verification at AWE

**Helen White, Aled Richings, Keir Allen, Attila Burjan, Sarah McOmish, Martin Morgan-Reading, Clive Pearman, Seb De Muynck and David M Chambers**
Atomic Weapons Establishment (AWE), UK



***Abstract:***

*Since 2000 AWE has carried out a research programme to investigate candidate technologies and methodologies that might support potential nuclear treaty verification regimes, in support of UK MoD objectives.*

*This paper will discuss AWE's current perception of the key technical problems in treaty verification. The challenges in key areas (i.e. device and material monitoring; chain of custody; authentication, certification of equipment and data; inspection studies and verification system studies) will be introduced, along with the proposed future programme of work aimed at addressing some these outstanding questions.*

**Keywords:** Nuclear Weapon, Dismantlement, Treaty Verification

## 1. Introduction

Since 2000 the UK Atomic Weapons Establishment (AWE), has supported a programme of research into technologies and methodologies that have potential to be used during the verification of possible future nuclear arms control agreements [1,2.3].

The programme looks at hypothetical bi-lateral or multi-lateral scenarios involving the dismantlement of nuclear warheads. The focus at this time is on a representative scenario which follows the nuclear warhead from the arrival at an initial storage facility, through various transport and dismantlement phases, through to the storage of fissile components. In this scenario, technical verification would be undertaken by an 'inspecting party' with reference to processes and facilities overseen by a 'host party'. Verification, in this context, is the process by which a level of confidence is reached in any declarations that have been made; ultimately this contributes to the judgement of whether the parties have adhered to the terms of any overarching agreement. The hypothetical declarations considered during this work contain a reference to a Treaty Accountable Item (TAI) (e.g. a complete warhead, a partially dismantled warhead, a fissile component, or fissile material from component destruction).

A key driver influencing the design of the final negotiated verification regime, and the selection of technologies, is the differing perspectives of the parties involved. This leads to the main challenge for any verification regime operating within a nuclear weapon complex: to provide the inspectors with the opportunity to gather sufficient evidence, whilst protecting sensitive or proliferative information held by the host. The final verification regime must balance these two viewpoints and take account of any resulting constraints that may be imposed.

The programme, up to this point, has supported a number of projects and technology trials investigating facility monitoring [2], device attribute measurements and information barriers [2,3,4], managed access [5,6] and chain of custody [7,8,9,10,11]. The work programme at AWE is split into five main areas: (1) inspection studies; (2) authentication, certification of equipment; (3) device and material monitoring; (4) chain of custody and (5) verification system studies. This paper discusses the key issues associated with each of these areas, overviews work within this programme to date and proposed future areas of work.

## 2. Inspection Studies

### 2.1. Introduction

The overarching bi-lateral or multi-lateral agreement drives the verification objectives of the inspecting party. These objectives, in turn, determine the activities the inspecting party will wish to undertake within potentially sensitive areas. The host party will assess a request for facility access to indentify the risks associated (a) with regards the potential release of sensitive and proliferative information and (b) with respect to safety regulations. The host party will then look to agree a set of 'managed access' techniques and procedures designed to mitigate those risks and, ideally, provide the inspectors with the desired level of access. The objectives of the inspection studies programme are to:

- Understand the levels of access that could be requested and assess the associated security and safety risks.
- Develop supporting processes, procedures and mitigation to protect sensitive and proliferative information, and allow the inspection to be conducted safely and effectively.

### 2.2. Work to Date

The inspection studies programme has supported a number of exercises designed to investigate the use of managed access techniques and the effective deployment of inspection activities in a working environment [3,5,6]. The managed access implementation associated with these scenarios can be intrusive from a host party perspective and restrictive from an inspector perspective [6]. Nuclear weapon complexes are highly regulated in terms of safety and security. Host concerns are not only for the TAI but also for sensitive physical security features associated with the facility itself. Should that facility have a dual remit, the host will also be looking to protect operations outside the boundaries of the inspection process. This could lead to access restrictions which limit the inspector's ability to fully assess a facility for diversion routes [6,11]; there may also be parts of the dismantlement process which the inspecting party cannot witness [11].

Typical managed access techniques which might be deployed by the host party might include searches, removal of prohibited items, identity checks, shrouding, exclusion zones, escorting, change barriers, host provision and operation of inspection equipment and host review of all data including personal notes [3,5,6]. The level of managed access may vary depending on the activities being undertaken by the inspecting party. This was demonstrated within the UK-Norway Initiative (UKNI) exercise series where there was an increase in managed access provisions between the familiarisation and monitoring visits [5,6].The inclusion of a TAI in the monitoring visit scenario, along with deployment of monitoring techniques, increased the risk of information release and therefore the host increased the constraints imposed to address the new threat [5,6].

### 2.3. Future Work

Following the 2010 'focused' exercise, undertaken within facilities overseen by AWE, the UKNI commented that a collaborative environment, and a proactive host, could help to facilitate the inspection process and increase confidence levels in the overall verification regime [6]. The level of managed access that has been deployed during these exercises was necessary, from the viewpoint of the host, but it can be difficult to explain that necessity in a scenario where elements of the process cannot be discussed [3,5,6]. Going forward, the Inspection Studies programme will look at lessons learned from other regimes in order to better understand what requests for access might be made and inspector 'expectations' in each case. The programme will also look to clarify the background requirement for given managed access approaches, thereby providing the underpinning understanding to be able to effectively design, and clearly communicate, managed access concepts.

## 3. Authentication, Certification of Equipment

### 3.1. Introduction

The deployment of equipment and the collection of data must be carefully managed to ensure mutual host and inspector confidence in the outputs of the regime.

Certification is the process undertaken by the host to ensure that all equipment [10]:

- Conforms to an agreed specification.
- Is incapable of communicating sensitive or proliferative information.

- Is compliant with facility safety regulations.

Authentication is the process undertaken by the inspectors to ensure that any data gathered [10,14]:

- Is an accurate record of the verification task or measurement.
- Is a true, unaltered output of a given instrument.

There is a likelihood that the host party will stipulate that all equipment must be 'host supplied' [15]. This adds an additional layer of confidence for the host that the supply of the equipment has been controlled, and that no link has been made between purchasing and the intended destination of the components. The host will then look to maintain control over the equipment during the certification process and deployment. There may be elements of the certification process that they do not want to reveal to the inspecting party. The inspectors will have similar goals, perhaps requesting that inspector supplied equipment be permitted, wishing to maintain control during the authentication process and having elements of the authentication process that they do not wish to reveal to the host party. To maintain dual party control under these circumstances is difficult. The differing perspectives of the host and inspecting parties mean that it is challenging to reach a negotiated compromise that satisfies both sides.

Authentication and certification processes are in addition to any system testing that might be performed as part of a manufacturing process. The approach to authentication and certification fall into four categories: assumption, association, direct and post deployment. Note that association and direct approaches might be undertaken at any point during the design, manufacturing or deployment process.

- Should the parties choose not to undertake any additional authentication or certification processes, any confidence in the equipment is based on the 'assumption' that it meets the criteria set out above. This is not a desirable approach as this leads to a lack of confidence in the deployed equipment and, therefore, the verification regime.
- 'Direct' concepts employ testing techniques on the equipment that is actually deployed. This would be the most robust approach, however, the issue of dual control, introduced above, means that the opportunities for deploying direct concepts can be limited. A combination of approaches might ultimately be used, however the host is likely to favour a direct certification as this gives the highest confidence that no proliferative or sensitive information could be released.
- The 'association' approach requires multiple copies of certified equipment to be provided by the host. A random selection process is then used to separate the copies into two sample sets. One of the sets is then released to the inspecting party for authentication; this equipment will not be deployed within the facility. The second set, to be deployed in the facility, is kept under dual control. It is the association between these two sample sets that provides mutual confidence in that deployment. Inspectors would prefer direct authentication, but in the scenarios played within UKNI, for example, authentication by association is often the negotiated compromise to satisfy both parties [5,6]. Again, this was in response to the need for dual control during the certification/authentication process and the host's insistence on direct certification [5,6]. This compromise does not necessarily provide the inspecting party with the same level of confidence as the host.
- 'Post' concepts rely on the release of equipment from the regime once the inspection process has finished, and the deployment of techniques to provide confidence that the equipment functioned as expected. Post deployment authentication as an option on its own is not generally favoured as any issues will only be detected after the regime has ended. In addition, in this scenario, the host is unlikely to release equipment from the facility once it has been deployed, which potentially restricts post authentication activities to those that can be performed within the facility. However, used in conjunction with earlier direct or association approaches, post authentication can provide additional confidence to the inspecting party.

### 3.2. Work to Date

In 2007 the UK Norway Initiative (UKNI) began a project focusing on the joint development of an item of inspection equipment [5]. The key aim of the joint technology development project is to understand how both parties might work together to ensure mutual confidence in the building and maintenance of inspection equipment. As a case study, the project has developed a gamma ray radiation attribute measurement system incorporating "information barrier" technology to prevent the release of sensitive

or proliferative information [4]; this project will be presented separately during this meeting [12]. In addition, the programme has also investigated novel information barrier designs, via a joint project with Aberdeen University on the potential application of artificial neural networks [3,13].

### 3.3. Future Work

Both the device and material monitoring and chain of custody programmes have highlighted authentication and certification as a key issue when considering which technologies to select for use in a regime. To date both areas have favoured simpler technology solutions, knowing from past experience that complex electronics and software could be onerous to authenticate and certify. This programme of work will look at the process of joint design and build with a view to clarifying how the processes of certification and authentication can interact to result in mutually trusted equipment. The programme will also evaluate techniques, methods and technologies required to undertake certification and authentication with a view to creating a range of supporting tools. Ultimately this will feed into the other areas of the arms control allowing for a broader range of technologies to be considered.

## 4. Device and Material Monitoring

### 4.1. Introduction

Inspectors overseeing the dismantlement of nuclear warheads will be looking to obtain a level of confidence that the object presented for dismantlement is as described in any given declaration. It is likely that technologies could be deployed which measure properties associated with the item, looking for the presence of fissile material and explosives in a viable configuration. However, the verification process must also account for obligations, on both the host and inspecting parties, to protect sensitive and proliferative information. For example, information related to masses, isotopics and configuration might be considered to be sensitive. This will not only impact on the design of measurement technologies, but also on the amount of information that can be revealed within any declaration. It should be noted that it is unlikely that the inspectors would be allowed to see the components, and are more likely to be presented with containerised objects.

The device and material monitoring programme:

1. Identifies attributes that would allow for the positive verification measurement to be made whilst protecting national security and proliferative information.
2. Evaluates technologies which might be used to verify declared attributes of a TAI.
3. Explores techniques and methods for protecting sensitive and proliferative information during the monitoring process.
4. Studies the impact of potential content of declarations, related to the TAI, which might be made in a treaty scenario.

### 4.2. Work to Date

Since 2000, AWE has undertaken diagnostic experiments of both nuclear and explosive materials to assess a variety of non-destructive, radiometric technologies in terms of (a) their ability to determine declared attributes and (b) the potential for the release of sensitive information [2]. The approach focuses on the fundamental physics of the problem by considering generic ‘material in a box’ scenarios. This allows for the investigation of key issues relating to the accuracy and veracity of given measurement techniques. This is particularly challenging in scenarios where the material configuration and container construction are unknown parameters, as could be the case for inspectors in a warhead dismantlement regime.

### 4.3. Future Work

Future work in this area will look at the efficacy of given attributes and the protection of sensitive information. This will improve understanding of declarations that can be made and the potential limits on measurement activities. This area requires multiple technological approaches, deployed to measure attributes at specified points during the dismantlement process. AWE will continue to investigate technologies to support a device and material monitoring regime. It should be noted that it

is unlikely that a monitoring regime will provide 100% proof but rather a level of confidence in the contents of any given declaration.

## 5. Chain of Custody

### 5.1. Introduction

Chain of custody is an integrated series of procedures and technologies designed to provide:

- Access control over Treaty Accountable Items (TAIs).
- Control over inspection equipment and data.
- Confidence that no attempt has been made to tamper with, spoof or divert TAIs, inspection equipment or data.
- The link between device and material monitoring measurements.

Multiple technologies are required to provide a flexible response to different scenarios, multiple levels of defence/evidence and to avoid the vulnerability of a single point of failure. As science advances, and potentially creates more sophisticated modes of attack, existing technologies must be reviewed and additional technologies considered. It is therefore clear that no single technology that can solve the chain of custody 'problem' [7,8,9,10,11]. With this in mind, the chain of custody programme has the following long term aims to:

- Develop a 'Toolkit' of technologies and procedures to support the development of various chain of custody regimes.
- Develop supporting technologies, techniques and procedures to enable the Host or Inspecting Party to effectively design, assess and deploy a chain of custody regime.

### 5.2. Work to Date

The current chain of custody programme began in 2008, and was primarily focused on understanding existing seal and unique identification technology developed in support of US/Russia treaties and IAEA Safeguards. Through 2008 and 2009, the team had the opportunity to become involved in a series of exercises including those instigated under the UKNI programme [3,5,6]. The lessons learned indicated that technologies aimed at boundary control would decrease the reliance on seals, provide boundary tamper indication and speed up the inspection process. This led to the instigation of new areas of work encompassing Tamper Indicating Enclosures (TIEs) [7,8], CCTV camera systems for boundary control [11], unique identification [10] and the use of non destructive evaluation (NDE) methods as either 'one-to-one' templates [9] or boundary inspection techniques.

In 2009 the chain of custody programme developed existing technical links with laboratories in the US (in particular Pacific Northwest National Laboratory and Savannah River National Laboratory). This collaboration covers a number of chain of custody concepts including the development of TIEs. Under the remit of the collaboration, two TIE technologies and a number of chain of custody systems have been investigated. The fibre optic and PMMA based TIEs have been reported previously [7,8]; the 'room-within-a-room' concept will be presented separately during this meeting [11].

### 5.3. Future Work

The above concepts have been the subject of a series of technology trials which have highlighted a number of research gaps; the forward programme is seeking to address these gaps with projects in the following areas:

- Deployment Strategy and Methodology: Design, develop and assess regime level methodologies to support the effective deployment of multi layer chain of custody regimes. This includes a regime level assessment of threats and vulnerabilities.
- Authentication, Certification and Inspection Methodology: Assess the potential vulnerabilities of chain of custody technologies/procedures; develop supporting authentication, certification and inspection techniques; and improve data management.
- Active Boundary Control: Develop a portfolio of technologies and procedures that would be used to provide continuous (active) monitoring of a controlled boundary or enclosure. An

active system is described as being powered, and gathering data, even when the system is left unattended by the Inspecting Party.

- Passive Boundary Control: Develop a portfolio of technologies and procedures that would be used to provide passive monitoring of a controlled boundary or enclosure. A passive system is either un-powered, or only powered when the inspectors are present to activate or check the integrity of the system.
- 'One-to-One' Templating Techniques: Develop a portfolio of technologies and procedures that would be used to measure a signature, associated with a container/object system, and use that signature to maintain chain of custody during transport or storage.
- Spoofing and Diversion Assessment Tools: Develop a portfolio of technologies and procedures that would be used to guard against spoofing/diversion activities associated with personnel, facilities, containers, equipment and shrouded items.

The chain of custody programme is looking to build on the research areas that are currently being investigated. In addition, the programme seeks to pursue new projects that can contribute to each of the above areas. With this in mind, the programme is currently establishing new links with academia which will support the current focus on passive boundary control and, ultimately, the newer areas of research. This programme recognises the importance of developing a balanced core toolkit of technologies to call upon, but also looks to support research which will ultimately allow technology providers to respond effectively to potential future treaty scenarios whose characteristics have yet to be defined.

## 6. Verification System Studies

### 6.1. Introduction

Device and material monitoring, chain of custody, authentication, certification, data management and managed access need to be brought together, and balanced, to create a mutually acceptable level of confidence in the overall verification system. As 100% confidence is unlikely, this area of work is investigating what levels of overall technical confidence might be achievable and how individual technical measures might be combined to maximise confidence in the regime. Verification system studies consider the whole process in order to understand:

- The overarching remits and objectives driving technical verification activities within a nuclear arms control scenarios.
- The use of definitions and terminology to underpin clear communication.
- The role of 'trust' and 'confidence' in a given verification system.
- How the above areas interact to create an overall judgement of technical confidence.
- Modelling and design tools to facilitate best practice in verification system design.

### 6.2. Future Work

This is an emerging area of work for AWE. The exercise programme to date has given a flavour of how a verification system might be assessed and the concept of host-inspector 'confidence' or 'trust' in given regime [3,5,6]. The work so far has shown that the role of 'confidence' and 'trust' in a given verification system requires greater consideration. A new project for the UKNI, in collaboration with King's College, London, will investigate the theoretical framework for methodically measuring levels of 'confidence' and the subjective levels of 'trust' when assessing the outcome of the technical verification element of an arms control inspection.

## 7. Summary

This paper provided an overview of the five main areas of research undertaken by AWE into nuclear arms control verification: (1) inspection studies; (2) authentication, certification of equipment; (3) device and material monitoring; (4) chain of custody and (5) verification system studies.

To achieve a high confidence technical verification regime will take considerable underpinning research, of which the AWE programme can only be a small part. Significant additional research by

other organisations is required across all the five areas discussed in this paper to address some of the topics currently perceived as key areas for progression. These include:

(1) Inspection approaches in heavily regulated environments where host sensitivities result in a highly restrictive level of managed access.
(2) New techniques and approaches to equipment authentication taking into account (a) the likelihood of host supply and (b) requirement for both parties to maintain dual control during the certification/authentication process.
(3) The accuracy of attribute measurements (such as isotopics and masses for nuclear material and elemental composition for explosives), particularly in situations where the configuration of the material and the construction of the container are unknown parameters.
(4) New and novel techniques to provide boundary control as part of a chain of custody supporting a verified nuclear dismantlement regime.
(5) Understanding the role of 'trust' and 'confidence' in a technical verification regime.

The AWE programme encourages contributions from the wider community in order to achieve the ultimate objective of an effective nuclear weapon dismantlement verification regime.

## 8. Acknowledgements

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# State Regulatory Authority (SRA) Coordination of Safety, Security, and Safeguards of Nuclear Facilities: A Framework for Analysis

**Stephen Mladineo [1], Sarah Frazar [1], Andrew Kurzrok [1]; Elina Martikka [2], Tapani Hack [2], Timo Wiander [2]**

[1] Pacific Northwest National Laboratory
P.O. Box 999, Richland, Washington 99352 USA
Email: Steve.Mladineo@pnnl.gov
[2] Radiation and Nuclear Safety Authority, STUK, Finland

***Abstract:***

*In November 2012 the International Atomic Energy Agency (IAEA) sponsored a Technical Meeting on the Interfaces and Synergies in Safety, Security, and Safeguards for the Development of a Nuclear Power Program. The goal of the meeting was to explore whether and how safeguards, safety, and security systems could be coordinated or integrated to support more effective and efficient nonproliferation infrastructures. While no clear consensus emerged, participants identified practical challenges to and opportunities for integrating the three disciplines' regulations and implementation activities. Simultaneously, participants also recognized that independent implementation of safeguards, safety, and security systems may be more effective or efficient at times. This paper will explore the development of a framework for conducting an assessment of safety-security-safeguards integration within a State. The goal is to examine State regulatory structures to identify conflicts and gaps that hinder management of the three disciplines at nuclear facilities. Such an analysis could be performed by a State Regulatory Authority (SRA) to provide a self-assessment or as part of technical cooperation with either a newcomer State, or to a State with a fully developed SRA.*

**Keywords:** 3S; facility operations; regulatory development

## Introduction

It has been long recognized that there are organizational and technical incompatibilities and at the same time overlaps among Safety-Security-Safeguards (3S)[1] requirements in nuclear facilities. In 2008, led by Japan, the G8 countries agreed to support the concept of 3S. The objective was to set up nuclear energy infrastructures in countries that were beginning nuclear programs. The idea was to help countries to integrate their approach to implementing safety, security, and safeguards measures so that they all receive appropriate attention, and through regulatory development, training. This approach would enable countries to take advantage of the synergies among the 3S components, while recognizing the differences.

Subsequently there has been much discussion about the concept of 3S. ESARDA and INMM meetings have devoted sessions to the topic.

[1] Acknowledging that there is no IAEA definition for 3S, we use the term in this paper as shorthand for Safety-Security-Safeguards.

In July 2012 at the Fourth International Meeting on Next Generation Safeguards Implementing Comprehensive Safeguards Agreements and Additional Protocols in Hanoi there was considerable discussion of the concept in breakout sessions. In November 2012 the IAEA conducted a Technical Meeting in Vienna on Safety, Security and Safeguards: Interfaces and Synergies for the Development of a Nuclear Power Programme. In the United States, the National Nuclear Security Administration has commissioned a study of the concept. There will be a special session on the topic at this summer's INMM meeting in Palm Desert.

These discussions about 3S are typically conducted at a high level and rarely examine specific examples of when safeguards, safety and security measures intersect. In these cases, the reviews—while moving the literature forward—are typically limited to one element of 3S such as non-destructive testing [1], probabilistic risk analysis [2], or operator organizational structure [3]. Consequently, these discussions do not examine thoroughly whether such intersections reinforce or

undermine the objectives of each discipline. Indeed, existing literature on the topic of 3S appears to echo the assumption that these intersections are inherently beneficial to the operator. The literature often continues to argue for greater coordination and integration of the three disciplines, particularly during the nuclear infrastructure development process.

The assumption that 3S is inherently beneficial does not encourage consideration of fundamental questions: Is there really a need for greater coordination and integration among the disciplines? Are there really a lot of conflicts among the disciplines? Would greater coordination and integration have a measureable impact on facility operations or reduce perceived vulnerabilities or risks?

When we looked for examples that required a coordinated 3S intervention, we found a few, but each seemed to have reasonably simple solutions. The classic example has to do with emergency exits. You need to let people crash out of a facility in a fire emergency, but there is also a need to subsequently corral them for security reasons. There is also a safeguards component to this example that may require a physical inventory verification prior to letting people back into the facility. This well-known example seems to persuade people that there is something important about 3S, but more examples are needed to warrant promotion of the concept. Some other areas ripe for research and development include access control, facility design, and computer security risks.

Without a thorough examination of the intersections among the three disciplines, nuclear plant operators and regulators are limited in their conclusions about whether and in what contexts greater coordination among the disciplines would truly generate beneficial results. In other words, despite the 3S discussions that have taken place, there has been little rigorous analysis of the merits and liabilities of the 3S concept. The purpose of this paper is to propose a framework for analysis of the concept, and then suggest how various entities might employ the framework.

## Framework

As part of the effort to develop a framework, we sought to record in a table intersections of safeguards, safety and security through the identification of discrete events that trigger interactions among the three disciplines.

Working from left to right, the table's four columns specify areas of impact ("Intersection Area"), the specific instigator of a possible 3S conflict ("Event"), the risk(s) posed by that event to one or more of the 3S disciplines ("Event Impact"), and potential ways to address these risks by restoring the equilibrium among the three disciplines ("Risk Mitigations"). By investigating each element of the framework individually, while focusing on the discrete events where the disciplines intersect, regulators, designers, operators and staff may be able to better identify who can take best advantage of or reinforce the 3S concept.

The table requires two prefatory notes:

When describing 3S intersections, risks and opportunities are two sides of the same coin. A situation can either be characterized as a risk that requires mitigation or an opportunity that overcomes a challenge. For example, the table could be structured to say either that "separate nuclear laws create conflicts and regulators should integrate them" or "an integrated nuclear law allows regulators to avoid conflicts that can arise". We have chosen for consistency to structure the table as risks that require 3S mitigation.

Further, it is important to note that this framework focuses on 3S impacts. As described in Suzuki et. al. [2], interactions between the three disciplines can include one or more of the three discipline pairs (safety-security; safety-safeguards; and security-safeguards) or impact all three disciplines. Table 1 is limited to the narrow set of interactions that force an intersection among all three disciplines, but future work could expand the table to capture additional events that only affect one or more of the discipline pairs.

| Intersection Area | Event | Event Impact | Risk Mitigations |
|---|---|---|---|
| **Nuclear laws and regulations** | Development of a national nuclear law | Legislation with multiple compliance- or performance-based regulatory requirements for each of the three disciplines may create conflicts for operators | An integrated law can reduce the risk of conflicts among the requirements imposed by each of the three disciplines. Further, regular self-assessments, benchmarking with other nations' laws and cross-training can mitigate risk of inconsistent or conflicting regulations. |
| | Creation of regulatory organization(s) and authority(ies) | Regulators (or offices within a single regulator) may experience friction over jurisdiction and prioritization. | With appropriate management attention, an integrated nuclear regulatory authority can minimize friction points and provide processes to resolve conflicts. |
| **Inspections** | State Regulatory Authority Inspection | Inspections focused on single disciplines might suggest recommendations that are detrimental to the other disciplines. Multiple inspections increase the burden on operators. | Cross training of inspectors. |
| **Emergency Preparedness & Response** | Evacuation emergency (e.g. fire, criticality) at a facility that requires rapid egress | Personnel safety takes priority, potentially undermining security of the plant or of nuclear material, and risks loss of containment/surveillance continuity of knowledge about the material. | Security layers (e.g. secure evacuation points inside and outside the facility or surveillance on emergency exits). Physical inventory taking prior to return to normal operations can determine that no material is missing. |
| **Information networks (computer security)** | Computer-enabled attack on facility infrastructure that threatens security systems or plant operational systems (e.g. control room) | Confidentiality, integrity, and availability of security, safety systems and safeguards information are all at risk. | Joint 3S vulnerability assessments of computer systems, potentially including penetration testing, fuzz testing, or exploiting data from one discipline to attack another can reveal potential points of attack and response solutions. |
| | Physical attack on facility's information network that allows unauthorized access to facility's | Confidentiality, integrity, and availability of security, safety systems and safeguards information are all at risk. | Joint 3S vulnerability assessments of computer systems can reveal potential points of attack and response solutions. |

| | | | |
|---|---|---|---|
| | information systems | | |
| | Development of facility IT platform(s) [4] | Data stored for each discipline independently can become stove-piped, or be not readily comparable. | Virtual consolidation of data on one secure supervisory software IT platform may yield savings on IT infrastructure while potentially making it more reliable and resilient. |
| | Development of facility IT platform(s) [4] | Data from multiple disciplines stored on a single platform could be susceptible to single point failure, or could risk cross-discipline compromise. | Appropriate backup systems and role based access procedures can mitigate these risks. |
| | Communications protocols development [4] | Multiple protocols for similar tasks across the three disciplines (e.g., reports, alarms) can unnecessarily burden resources. | Implementation of standard communications protocols can yield savings on IT infrastructure, recognizing that as described by Stein et. al. [4], standard communications protocols would have separate authentication and encryption schemes for each user role. |
| **Training** | Inspector training, curriculum design | Inspectors trained to narrowly focus on their own discipline may miss seeing and reporting potential risks or vulnerabilities. | Cross-trained inspectors are more likely to identify potential risks or vulnerabilities in operational practices or regulation implementation; continual training helps keep all inspectors aware of new threats and points of consideration. |
| **Operations** | Diversion of nuclear material | An unauthorized diversion may create security and safety risks in addition to safeguards noncompliance. | IAEA safeguards inspections could trigger a safety or security alarm through a report of an anomaly. |
| | Inventory of nuclear material in storage | Conducting inventory can place safety (e.g., minimizing dose) in tension with safeguards and security (minimizing time that material is subject to theft or diversion). | Technical solutions could mitigate safety and security risks (e.g. remote verification, or containment and surveillance). |
| | Surveillance and assessment camera installation [3] | Independent safety, security, and safeguards cameras may be redundant and can strain resources. | Thoughtful placement of some cameras (with effective encryption and authentication for safeguards) could increase the facility's flexibility to respond to events in a defense-in-depth manner while reducing costs. It may be necessary to retain some non-networked cameras. |
| | Facility management process development [5] | The effectiveness of operations in each discipline can suffer from weak management. | Implementing advanced management concepts such as quality assurance, root cause analysis, equipment preventative maintenance, risk assessment, configuration management, design control, change control, document control, and records management reduce the risk of poor management oversight. Corporate governance policies, |

| | | | |
|---|---|---|---|
| | | | segregation of duties, identifying dangerous work combinations and good communications all play a significant role. Poor management can be a result of not knowing the big picture. |
| | Establishment of material accounting areas [3] | Gaps or overlaps in material accounting areas can introduce diversion opportunities and force duplicative equipment procurements. | Design accounting areas to avoid duplication of equipment and IT tracking conflicts. |
| | Facility design verification [3] | Facility as-constructed must meet safety, security, and safeguards design criteria, but developing verification procedures independently may waste resources or overlook vulnerabilities. | Planning validation and verification procedures to proceed with 3S in mind can resolve potential conflicts and save resources. |
| **Facility design** | Design of storage vaults | Opening the vault for inspection can create environmental dynamics that pose safety risks and weaken the security posture. | Tags/seals and unattended remote monitoring systems can maintain continuity of knowledge, meeting safeguards needs while mitigating the safety/security risks of an open door. |
| | Design of data center(s) | Three independent data storage systems can lead to waste on certain shared needs (e.g., back-up power generation, access control) and can introduce risk through inconsistency of information. | Jointly housing data could optimize facility usage and provide savings on installation and facility space, and minimize inconsistencies. Data backup is essential to avoid single point failures. |
| | Risk Assessment [2] | Risk assessment methodologies within each discipline that do not consider the impact on other disciplines can create design conflicts and introduce risk. | Ensuring that risk assessment is harmonized across the three disciplines encourages optimized design choices. |
| | Design of nuclear safety containment system [3] | A narrowly-focused nuclear safety containment design can create security or safeguards risks. | 3S-cognizant nuclear safety containment design simultaneously reduces risks from accidents, theft, or diversion. |
| | Definition of physical protection system requirements [3] | Physical protection requirements impact radiation protection, loss avoidance, and safeguardability. | Developing facilities with 3S in mind reduces the risk of design conflicts and allows designers to optimize resource usage. |

**Table 1:** 3S Events, Impacts, and Risk Mitigations

## Preliminary Conclusions

From our preliminary findings suggested in the table, we can assert that there is value in the concept of 3S, and that a framework can be developed to support thoughtful analysis about how the three disciplines interact at a facility. Moreover, there are instances where safeguards, safety and security measures could be better coordinated, integrated or considered holistically by regulators, operators, designers and staff to improve facility operations. Recognizing that our framework and approach are not complete, we provide recommendations for various users to advance discussion on this topic. More research will be needed to validate our preliminary conclusions.

## Implications

### Regulators

Each State will choose its own approach to handling the coordination or integration of the three disciplines. Responsibilities for each discipline may reside in one or more organizations. In a nuclear newcomer state, it may be important to keep all of the issues under one organization to ensure that none of them is ignored. In a state with an advanced nuclear energy infrastructure, the disciplines may need to be managed separately. A regulatory authority with responsibility for all the disciplines may be most cost effective. A State with more than one regulatory authority will need to closely coordinate among these authorities to avoid conflicts or gaps. Either approach can ensure that regulators are able to see possible contradictions among requirements for operators. The regulators for each of the three disciplines should be cognizant of the requirements of the other disciplines. With cross training of inspectors, they will be able to recognize conflicts when inspecting for any of the three disciplines. This would ensure that harmonization of regulations will prevent conflicts among requirements for each discipline.

### Operators

In an operating facility, nuclear safety can eclipse nuclear security and safeguards due to its direct impact on worker safety and health and facility operations. Because of its overriding priority, safety will often drive the budget, limiting the amount of money dedicated to safeguards and security measures. There are consequences to allowing this to happen. Reducing security measures can leave vulnerabilities. Limiting resources for safeguards implementation can lead to inaccurate or incomplete reporting to the International Atomic Energy Agency, which could generate suspicions that there has been a diversion. These suspicions may require greater IAEA scrutiny and intrusion into facility operations. A rigorous assessment by the operator of how the three disciplines benefit most from harmonization, or how they are likely to generate vulnerabilities would help inform resource allocation for a facility.

### Designers

For nuclear facility design engineers, safety has been the overriding paradigm. Accidents at Three Mile Island, Chernobyl, and Fukushima have reinforced that concern. Industrial safety, criticality safety, radiation safety, electrical safety, high pressure safety, and other areas of safety are likely to be second nature to the designers. In the past, security was left to the security professionals, and therefore was often an add-on to the facility design and computer systems. International Safeguards has had even less emphasis in the design community, and has had to be developed as an afterthought.

One example is the first final-disposal facility for spent nuclear fuel, currently under construction in Finland. The facility will consist of an encapsulation plant and a geological repository for spent nuclear fuel.
The safety of final disposal is a fundamental issue. The current process and plans for the encapsulation plant and geological repository have been designed primarily with safety in mind. The implementation of security and safeguards must not compromise safety. However, challenges appear from the security point of view. These include the identification of vital and inner areas; creation of a new design basis threat (DBT); identification of interfaces, synergies and challenges between safety, security and safeguards, and handling classified information throughout the lifecycle of the facility.

The safeguards concerns of final disposal are unprecedented. Traditional safeguards measures for the spent fuel will be impossible once the fuel is stored and tunnels are backfilled. The whole process must provide sufficient assurance that the fuel has been stored and emplaced according to declarations and that no diversion has been possible.

Several safeguards decisions are part of broader 3S considerations. For example:

- The calibration of the equipment used for verifying spent fuel before encapsulation involves both computer security and safeguards aspects,
- Continuity of knowledge from the first verification of spent fuel to the last one before encapsulation is an issue that involves safety, security and safeguards.
- Once the capsules are moved from the encapsulation plant to the final disposal storage underground verification is no longer possible. This movement involves safety, security and safeguards.

Some challenges have occurred in the design of the facility, underscoring the importance of recognizing the safeguards requirements early in the design phase. An analogy is designing a house without considering electricity, and it is only once construction is done that you realize that you forgot to include cabling. In a nuclear facility, particularly one where the operating duration is measured in centuries, including safeguards in the design phase would prevent such problems. Further, during the life of the facility some parts may be storing nuclear material while other parts will still be under construction. Therefore, it is necessary that safeguards be designed into the facility along with safety and security to manage these activities.

The US National Nuclear Security Administration's Next Generation Safeguards Initiative has sponsored development of a series of Safeguards by Design guidance documents to be used for reference by designers [6]. Consideration of all three disciplines during the design phase may lead to operational efficiencies but more research is needed in this area.

### Staff

In previous articles some of the current authors have discussed the culture of the three disciplines (safety-security-safeguards) using an organizational culture construct [7]. We argued that safety culture was the best developed of the three, with nuclear security still catching up since the attacks of September 11, 2001. Safeguards culture is the least developed of the three, partly because there is no natural analogue to international safeguards in daily life. Additionally, safeguards culture does not benefit from having an internationally agreed-upon definition as do safety culture and nuclear security culture. Even less developed is the concept of a 3S culture. (We have argued elsewhere that an international agreed-upon definition of safeguards culture should be a next step prior to consideration of 3S culture.) [8] The framework developed in this paper suggests there are specific actions staff may be able to take to reinforce 3S objectives and communicate the importance of all three disciplines in an organization. A closer examination of these actions could lead to more productive development of the 3S concept.

## Conclusions

We have proposed a framework for analysis, but have not performed the rigorous analysis that we argue needs to be done. Rather than continuing to talk about 3S in the abstract, we propose a focused research project that brings together experts from each of the disciplines to give the nuclear community a better understanding of the range of the set of specific conflicts that 3S integration could help resolve. Once understood, communities of regulators, operators, and designers can begin to discuss best practice. By taking actionable steps to scope and implement 3S, the international community can take advantage of the potential benefits of 3S activities.

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# Intrinsic fingerprints inspection for identification of dry fuel storage casks

**Dmitry Demyanuk[1], Michael Kroening[1], Andrey Lider[1], Dmitry Chumak[2], Dmitry Sednev[1]**

[1] National Research Tomsk Polytechnic University, Lenin Avenue, 30, Tomsk, 634050, Russia
[2] Odessa National I.I. Mechnikov University, Dvoryanskaya str., 2, Odessa, 65082, Ukraine
E-mail: sednev@tpu.ru

***Abstract:***

*MIT study on the Future of Nuclear Fuel Cycle has showed that spent nuclear fuel (SNF) is a significant potential source of energy. The energy content of SNF is significant and thus there is incentive to maintain the option of future use of SNF. A lot of technologies are applied for SNF storage. One of them is a Dry intermediate storage (DIS). Choosing a DIS system technology is not an easy choice: the decision involves a combination of technological, political, licensing, and policies parameter. However, it gives a lot of advantages to potential user: efficient management of SNF at the site; passive & inherently safe design; accountability and effective safeguards. In other words, it meets the idea of 3S: safety, security and safeguards synergy, all widely discussed at the Seoul summit. The idea was aimed at improving the efficiency of the cooperation between these elements, which have relatively large number of common points, but at the same time have their proper features. Careful consideration has showed that the sphere of non-destructive testing (NDT) is most appropriate one to start 3S synergy implementation. NDT methods and devices are widely applied in each S-element and could quite often be retargeted. Therefore, in this paper we describe our experiments in using information, which can be collected during common ultrasonic inspection of DIS cask seam weld for safety reasons. We process the data and use it for security and safeguard purposes as information containing unique intrinsic fingerprints of the cask's material.*

**Keywords:** Intrinsic fingerprints**;** spent nuclear fuel; accounting; dry fuel storage; ultrasonic inspections.

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## 1. Background

The International Atomic Energy Agency (IAEA) Director General Yukiya Amano, in his statement to Fifty-Sixth Regular Session of IAEA General Conference in 2012, emphasized that "…Eighteen months after the accident [Fukushima Daiichi accident], it is clear that nuclear energy will remain an important option for many countries. Our latest projections show a steady rise in the number of nuclear power plants in the world in the next 20 years"[1]. It means further dissemination of nuclear energy for peaceful use among new countries and in the new regions as well as improving existing capabilities.

Nuclear energy technology is complex and sophisticated and requires a high level of scientific development, but at the same time poses potential dangerous for humanity and requires the most advanced and well-thought actions. Despite the worldwide robust experience in operating nuclear facilities and related infrastructure, the Fukushima accident showed that the use of nuclear energy still can lead to disastrous consequences. Thus the system of nuclear energy utilization requires new and innovative ideas to improve the level of nuclear safety, nuclear security and nuclear safeguards. These elements create a stable system for the peaceful use of nuclear energy.

Within “2012 Seoul Nuclear Security Summit”, which held in Seoul in March 2012, as one year past after Fukushima Daiichi accident, synergy between safety and security was one of the main topics of discussion. On the official Seoul Summit Communiqué highlighted: “Acknowledging that safety measures and security measures have in common the aim of protecting human life and health and the environment, we affirm that nuclear security and nuclear safety measures should be designed, implemented and managed in nuclear facilities in a coherent and synergistic manner” [2]. Additionally to this statement, in the context of protecting human life and health and the environment, safeguards measures (non-proliferation) also need to be taken into account. Thus, today, the key challenge for scientists, politicians, diplomats is to seek options for preventing any situation which can undermine the peaceful use of nuclear energy. One of the ways of resolving this problem is to enhance the synergies between nuclear safety, security and safeguards.

Careful consideration has showed that the nondestructive testing (NDT) is one of most appropriate option to start 3S synergy implementation. NDT methods and devices are widely applied in each S-element and quite often could be retargeted. [3] To understand what should we choose as very first object for synergy implementation nuclear security means we try to screen for most vulnerable stages in the nuclear fuel cycle. According to official Russian ROSATOM guidelines to applying of sealing technologies: “...there is no official data concerning sealing systems that could not be faked...” [4].

Within 3S initiative Tomsk Open Laboratory for Material Inspections of Tomsk polytechnic university has launched the project on NDT methods application for fingerprinting by intrinsic features of dry intermediate storage (DIS) casks with spent nuclear fuel (SNF). Nowadays, NDT is mainly focused on safety purposes, but it seems possible to apply those methods in order to provide national and IAEA safeguards. [5] The containment of spent fuel in storage casks and vaults at an DIS could be dramatically improved if so-called “smart” spent fuel storage and transfer casks were developed. Such casks would have tamper indicating and monitoring/tracking features integrated directly into the cask design. These features could be an add-on package or be integrated into the cask construction. The microstructure of the containers material as well as of the dedicated weld seam applied to the lid and the cask body provide for a unique fingerprint of the full container, which can be reproducibly scanned using an appropriate technique. Weld seams investigation performed by ultrasonic backscattering technique for providing those fingerprints is described in the paper.

## 2. Overview of a dry cask storage design

The energy content of SNF is significant and thus the incentive to maintain the option of future use of SNF. The historical vision of the future of the nuclear fuel cycle was that LWR SNF is a valuable resource. Plutonium from LWR SNF was to be recovered and fabricated into fuel for the start-up of fast reactors. Such a system could increase the available energy from uranium by more than an order of magnitude. [6]

The technologies currently available for spent fuel storage fall into two categories, wet and dry, distinguished according to the cooling medium used. Whereas wet storage option has been used for spent fuel storage and cooling at reactor sites and in some off-site storage facilities around the world, a variety of technical methods for dry storage have been developed since then and are available in the international market.

Intact fuel storage, which is the most prevalent dry storage method, refers to storage of fuel assemblies with no attempts to pre-compact them or alter them by destructive methods prior to storage. A variety of storage systems have been developed to meet specific requirements of different reactor fuels and a number of designs based on these generic technologies are now available for dry storage for the spent fuel containers (also called casks) or vaults (horizontal, vertical etc). The technology continues to evolve keeping up with the design optimization and new materials. One of the driving forces of the trend towards dry storage options (especially those of casks) is the inherent technical flexibility linked to economics. Compared to the pool facilities, which need to be built at full capacity initially, the modular type dry facilities can be added as needed with the advantage of minimizing capital outlays. [7]

| Item | Type of Storage (ISFSI/AFR Facility) | Containment & Shielding | Features | Notes |
|---|---|---|---|---|
| 1 | **Independent Spent Fuel Storage Pool (Wet Storage of Spent Fuel)** | Water in a deep stainless steel lined concrete pool within an enclosed structure | Traditionally used for expanded storage of spent fuel Many examples worldwide; | Current trends are towards dry rather than wet storage, for reasons of reduced maintenance (water management) and improved safety and security. |
| 2 | **Vertical Dual-Purpose Spent Fuel Dry Storage/Transfer Cask** | Heavily shielded steel cask with spent fuel sealed in inner steel canister (i.e. double lidded) | Vertical, dual purpose, dry spent fuel storage and transfer cask | Examples: CASTOR, TN NAC-ST/STC BGN Solutions |
| 3 | **Vertical Concrete Dry Spent Fuel Storage Cask/Silo** | Heavily shielded concrete cask/silo with spent fuel sealed in an inner steel canister | Vertical, dry spent fuel storage cask/silo | Examples: CONSTOR HI-STORM |
| 4 | **Horizontal Modular Concrete Dry Spent Fuel Storage** | Heavily shielded modular concrete storage with spent fuel sealed in an inner steel canister | Horizontal, modular dry spent fuel storage vault | Examples: NUHOHMS NAC-MPC/UMS MAGNASTOR |
| 5 | **Concrete Dry Storage Vault with Thimble Tube Storage Wells** | Heavily shielded concrete vault with thimble tube storage wells for spent fuel | Vertical, dry spent fuel or vitrified waste storage vault with thimble tube storage wells | Examples: MVDS MACSTOR |
| 6 | **Dry Geologic Storage (Tunnel or Mine)** | Dry gas-filled spent fuel canisters emplaced in an isolated deep tunnel or mine and backfilled with earth | Dry spent fuel storage in tunnel or mine; vertical or horizontal fuel orientation | Example: Olkiluoto NPP Spent Fuel Repository (Onkalo) |

**Table 1:** Storage options for AFR storage of spent fuel [8]

Several variations of these fuel storage concepts, often by combination of existing dry storage technologies, have been developed with prospective applications in the future. Cask is currently the most popular option that can be purchased or leased from the competitive market for expedited installation, on the assumption that the necessary license can be obtainable and any other obstacles such as opposition from the affected local community have been or could be resolved. Inheriting the technology initially developed for large-scale transportation of spent fuel from storage to reprocessing operations, several large size casks are now being marketed for storage services. Concrete modules

have also become popular as a competitive option, with more designs licensed and implemented over the years. Markets for concrete modules are merging with those for vaults as a compact storage system, in terms of advantages when land availability is an issue. Multi-purpose technologies (i.e. a single canister package for storage, transportation and disposal) have also been developed, for instance in the US, although its use for disposal package is subject to uncertainty instance in the US, although its use for disposal package is subject to uncertainty. Within this paper we will focused on Vertical Dual-Purpose Spent Fuel Dry Storage/Transfer Cask as the most popular type of DIS cask in Russia.

## 3. Ultrasound inspection

The ultrasound testing (UT) method most commonly used for material inspection is the pulse-echo method. This method is an echo-sounder technique. The ultrasonic pulse is usually generated by piezo-electric conversion and is called transmitter pulse. Other conversion techniques like electro-magnetic conversion (EMAT) will not be considered in this paper. The pulse propagates through the material that we assume to be isotropic. Part of the acoustic wave is scattered back to the transducer and converted again by the reciprocal piezo-effect. The received pulse is called A-scan. The amplitudes of the A-scan indicate the intensity of reflection that in some way but not well defined correlates with the reflector dimension, the arrival time of reflector amplitudes is used for their localization assuming straight propagation along the center line of the wave field intensity profile. [9] Standard sensitivity setting for reflector detection and reporting references to measured amplitudes of defined calibration reflector geometries, and acceptance criteria are considered in addition length or area of localized acoustic reflections and in special cases the systematic appearance.

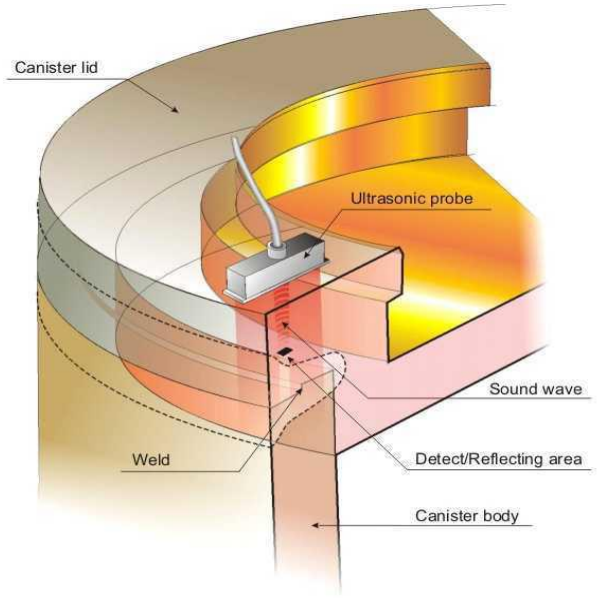


**Figure 1:** Typical UT inspection of cask weld

Figure 1 represent scheme of Vertical Dual-Purpose Spent Fuel Dry Storage/Transfer Cask UT inspection for safety reason. For the detection of structural features, especially planar, we need normal incidence of the wave field to receive the backscattered reflection. Depending on possible critical flaw geometries and a risk classification of the inspected component the inspection procedure requires various angles of incidence.

The required angle of incidence is realized by plastic wedges of different sound velocity and low attenuation. Snellius' law can be used for the wedge design of the specified angle of sound incidence as long as the beam is directed. A typical angle beam transducer design is shown in figure 2.

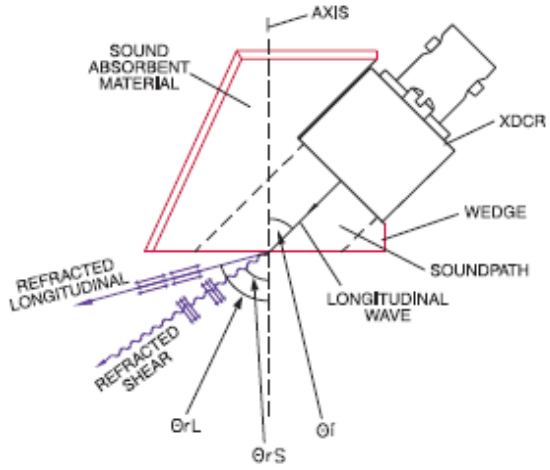


**Figure 2:** Typical design of an angular beam transducer

Both modes, the longitudinal mode (cL = 5920 mm/sec in steel) and the slower shear mode (cs = 3250 mm/sec in steel) can be realized by the appropriate wedge design. Above the first critical angle of 90° refracted longitudinal mode only shear waves are transmitted into the sample. We use acoustic noise received from both modes traveling through controlled material for structural fingerprinting purposes. Raleigh scattering forms most of the acoustic noise that is usually not evaluated for flaw detection and assessment. However, we assume that the acoustic noise depends on the microstructure that is specific for the position. Geometric extended reflectors are by far more position related and may be evaluated for position data and position changes [10]. However, in high quality steels as used for structures extended material flaws or material inhomogeneity are rather rare. Further, only point reflectors will provide accurate data as required. Back-wall reflection, for example, is independent of the transducer position, and compound reflector geometries, such as crack faces, require careful analysis of the radio frequency (RF) A-scans to separate individual and interfering scatter centres [11].

During the investigation the number of experiments was conducted. The main goal of experiments was the development of methodology for structural fingerprints identification. Experimental set-up was developed to achieve the goal. Russian universal industrial ultrasonic flaw detector USD 60 was chosen as basis for the installation. [12] (Fig.3) Angle beam transducers by Olympus Panametrics and standard laptop with software included in the set-up, also.


**Figure 3:** Experimental installation

Three steel plates with different thickness (8, 10, 12 mm) were welded and used as samples dedicated to be equivalent of cask weld seam. Weld seams at each sample were cut off for 40 mm form the edge. Nine points of interest were chosen for investigation.

So-called “Area of interest” is used in order to prevent internal or external influence on material structure, within frameworks of the experiment. It is internal volume that is placed for 2 mm above internal surface of the sample. (Fig.4)

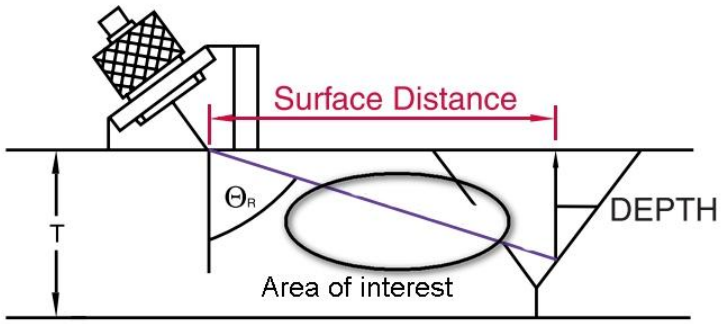


**Figure 4:** Scheme of sound distribution

In the investigation, we use standard software for data acquiring and processing. The development of data processing model includes experience from one of the most reliable identification systems at the moment – biometric one. Initial obtained data is image with graph (A-scan) that is why we chose the dactylography method as basis for the investigation. After taking a fingerprint as an image the system shall digitize and analyse the input data. Then the most distinctive point should be picked from a digitized image, and on basis of those points system will create a pattern for initial fingerprint. The same steps the system has to take for authentication process, but the last one will be matching points of patterns. So, if more than 65% of points are meet with each other, it will considered that identity of fingerprints was proven. [13] We took the 65% threshold for development and testing of the data processing system.

Ultrasound equipment allows us to use signals of different types. Radio frequency A-scan was chosen for the investigation, because it can present data in real scale like an oscillograph (Fig. 5)

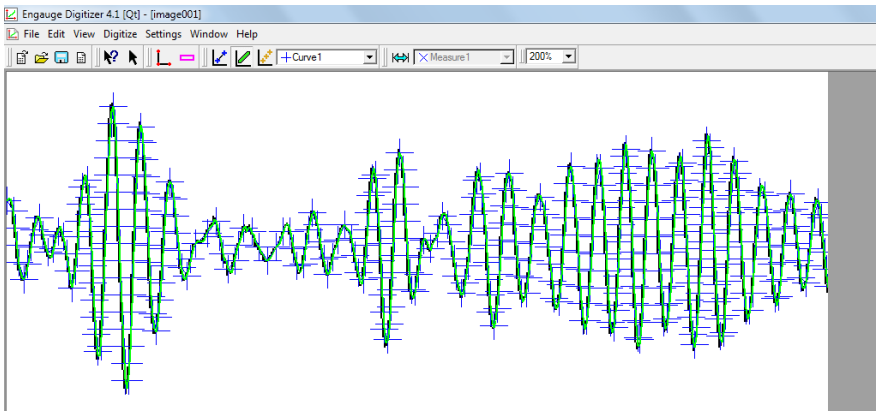


**Figure 5:** Data presented in radiofrequency A-scan

50 portable network graphics (png) images of the RF signal were acquired for each of nine points. Comparative data analysis proved the uniqueness of the acoustic noise image.

The initial images were digitized into a data arrays for further signal processing. All curves on images were described with X and Y coordinates. Number of points’ coordinates was averaged for mistake minimization. Then we found all peaks among the data array. Optimal quantity of peak was defined by empirical way. It is from 38 to 43. In framework of the investigation we consider to apply 40 peaks. (Fig. 6)

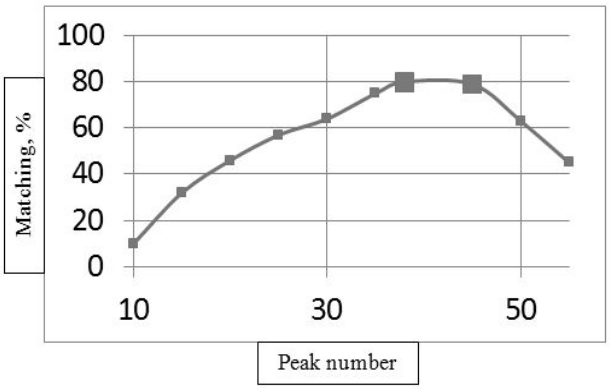


**Figure 6:** Data quantity optimizing

The obtained peaks are pattern for the chosen measurement point, and it describes is described with suitable function by means of software (OriginPro Lab 8.0). These data will be used for authentication. Inspector should make the control measurement in the chosen point, and then the system will match it with the pattern by dividing the taken function by the pattern one. The result will show difference between amplitudes in given X position. The confidence interval should be defined for evaluation of the obtained data. It helps to take into consideration the maximal number of the peaks and to minimize false data. Exceeding of the confidence interval range in 0.4 point will lead to significant increase of false data that taking into consideration. In this work the confidence interval will be in range [0,8 … 1,2].

In case of the successful authentication the match accuracy reaches from 72,5% to 87,5%. The divergences with the pattern are more often seen at the initial and final values of the data array. The data taken from the other measuring position shows the match accuracy within the interval 12,5% to 45%.

## 4. Conclusion

The result obtained in the experiment verified the consistency of the chosen combination of the control method and structural fingerprint appliance. In case of the successful authentication the levels of the most distinctive points' compliance exceed the given threshold on 65% which differs considerably from the percentage of the concurred points while authentication from the other points. Since the reproduction or doubling of the proposed unique identification characteristics is impossible at the current state-of-art, we can assume that the appliance of this technique will allow to identify the interference into the nuclear materials displacement with high accuracy.

Moreover, achieved result has shown that by only one measurement we can gather information for all 3S – ultrasonic inspection of DIS cask closure weld gave to us information about: possible cracks and flaws that we can use to evaluate degradation of material for safety reasons; intrinsic features of weld–natural fingerprint of material that can be used as non-tamperable seal for control and accounting purposes (security related issue); and last but not least safeguards on national and international level: automatic data acquisition and processing allow create data storage to make verification process easy and robust.

## 5. Legal matters

I agree that ESARDA may print my name/contact data/photograph/article in the ESARDA Bulletin/Symposium proceedings or any other ESARDA publications and when necessary for any other purposes connected with ESARDA activities.

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# Comparative assessment of the Pu content of MOX samples by different techniques

**R. Buda[1], R. Carlos-Marquez[1], E. Dahms[1], F. d'Amati[1], L. Emblico[1], J. Horta-Domenech[1], A. LeTerrier[1], K. Lützenkirchen[1], S. Millet[1], S. Morel[1], A. Mühleisen[1], A. Nicholl[1], V. Piron[1], D. Poublan[1], M. Ramos-Pascual[1], A. M. Sanchez-Hernandez[1], F. Sarli[1], P. Schwalbach[2], A. Terrasi[2], P. van Belle[1], M. Vargas-Zuniga[1], J. Zsigrai[1]*, E. Zuleger[1]**

[1]Institute for Transuranium Elements, Joint Research Centre, European Commission, Karlsruhe, Germany
[2]Directorate General for Energy, European Commission, Luxembourg
*E-mail: jozsef.zsigrai@ec.europa.eu

***Abstract:***

*The isotopic composition and concentration of Pu in eight "high-burn-up" mixed-oxide (MOX) fuel samples has been determined by destructive and non-destructive techniques. In addition, the U concentration and U isotopic composition was also available from the destructive techniques. The applied non-destructive techniques were gamma spectrometry, calorimetry and neutron coincidence counting, while the destructive techniques were titration, alpha spectrometry and thermal ionization mass spectrometry combined with isotope dilution. The current study describes the measurements and compares the results obtained by the mentioned techniques. Some lessons learned for the improvement of the non-destructive assay are also discussed.*

**Keywords:** High burn-up MOX, NDA, U and Pu assay

## 1. Introduction

In order to determine the Pu content of mixed-oxide (MOX) samples from fuel fabrication plants, safeguards inspectors usually send the samples to an analytical laboratory. The very precise measurement results obtained by destructive analysis (DA) techniques in the analytical laboratory are essential for safeguarding the fuel fabrication plants. However, the transport of the nuclear samples is usually quite difficult, involves lot of administration and it is expensive. With all this, the transport takes a long time, so it could happen that even more than a year passes before the analytical results are available.

For this reason it would be advantageous to do at least some of the measurements on-site, possibly without generating radioactive waste, to have a quick assessment of the Pu content of the verified materials before they are shipped to a laboratory for more detailed analysis. Non-destructive (NDA) measurement techniques make this possible.

In this work we investigated the performance of gamma spectrometry, calorimetry and neutron coincidence counting. Gamma spectrometry was used for determining the Pu isotopic composition, while calorimetry and neutron coincidence counting were used to measure the Pu concentration. The results from the non-destructive techniques were compared to results from isotope dilution mass spectrometry, which was taken as reference, and to results from potentiometric titration. A similar comparison involving some of the mentioned techniques has been done previously by Mayer et al. [1].

In section 2 we describe how the Pu isotopic composition was measured by gamma spectrometry and we compare the results to the result from thermal ionization mass spectrometry (TIMS). In section 3 the measurement of the Pu concentration by the various NDA and DA techniques is described. In

section 4 the results from the different techniques are compared, while in section 5 we give the conclusions and the description of possible further studies.

## 2. Determining the isotopic composition

Many of the techniques used for measuring the U and Pu content of the samples require the knowledge of the isotopic composition in order to be able to convert the measured signal to total Pu (or U) concentration. In addition, the isotopic composition itself is an important piece of information in the safeguards verification of nuclear activities.

In this work the non-destructive determination of the Pu isotopic composition of the investigated MOX samples was done by gamma spectrometry. The destructive option for measuring isotopic composition was thermal ionisation mass spectrometry (TIMS) combined with alpha spectrometry. The destructive analysis provided both the U and Pu isotopic composition.

Two high-resolution gamma spectrometers based on planar HPGe detectors were deployed. We call the two spectrometers “internal” and “external”, respectively. The “internal” gamma spectrometer was integrated with a neutron coincidence counter attached to a glove-box, so that a gamma measurement could be done simultaneously with a neutron measurement. This was the so-called "N-Gamma counter" or "OSL counter" ([2]-[5]). The “external” gamma spectrometer was based on a stand-alone HPGe detector, with its own lead shielding.

For the gamma measurements outside the glovebox, one can adjust the sample-to-detector distance and one can choose the absorbers between the detector and the source. We used 0.5mm of Sn foil in front of the samples and we chose the source-to-detector distance in such a way that the dead-time was around 1-2 % and the total count-rate was around 500-1000 cps.

For the gamma measurements in the N-gamma counter attached to a glovebox, in most cases, the Cd sleeve into which the samples where placed provided just the right filtering of the low-energy gammas from $^{241}$Am. In addition, by means of a cylindrical Al placeholder placed into the Cd-sleeve, the pellet samples could be positioned inside the counter very accurately. However, for one powder sample (see sample 17891 below) which contained almost 2 g of Pu, the filtering by the Cd sleeve was not sufficient and the dead time was quite high (12 %).

The gamma spectra from the two spectrometers were evaluated using the computer code MGA v9.7.

As $^{242}$Pu cannot be directly measured by gamma spectrometry the MGA code has to estimate it based on an isotopic correlation. The parameters used in the formula which relates the $^{242}$Pu content to the other Pu isotopes depend also on the isotopic composition. However, a simple empirical rule, based on calculating and indicator, exists for choosing the right parameter set [6], [7]. In this work the following correlation was used

$$\frac{^{242}Pu}{^{239}Pu} = C_0 \left[ \frac{^{238}Pu}{^{239}Pu} \right]^{C_1} \left[ \frac{^{240}Pu}{^{239}Pu} \right]^{C_2}, \qquad (1)$$

where we have chosen $C_0$=1.333, $C_1$=0.330 and $C_2$=1.700. Consequently, we have set the first three $^{242}$Pu parameters in the setup file for MGA accordingly, while the rest of the parameters were left at zero.

The investigated MOX samples contained "high-burn-up" Pu, with relatively high $^{242}$Pu (7%) and low $^{239}$Pu (~55%) fraction. This presents a challenge for gamma spectrometry and in particular for the MGA code, because it relies on the presence of the $^{239}$Pu gamma peaks, and because the estimation of the $^{242}$Pu fractions is more significant, than for low burn-up Pu. Nevertheless, the obtained results show that MGA V9.7 is able to cope with this situation.

Taking the TIMS results as reference, the performance of the two gamma spectrometers can be compared. As it can be seen from Figure 1, the Pu isotope abundances calculated from the gamma

spectra taken by the two detectors agree very well. Furthermore, the bias between the values from gamma spectrometry and values from TIMS is less than the reported uncertainty, for both spectrometers. However, taking a closer look at the graphs, one can see that the results from the "internal" gamma spectrometer seem to be worse, in general, than the results from the "external" gamma spectrometer, i.e. they deviate more from the TIMS results. This is also confirmed by the numbers in Table 1, where the average biases from the two detectors are compared.

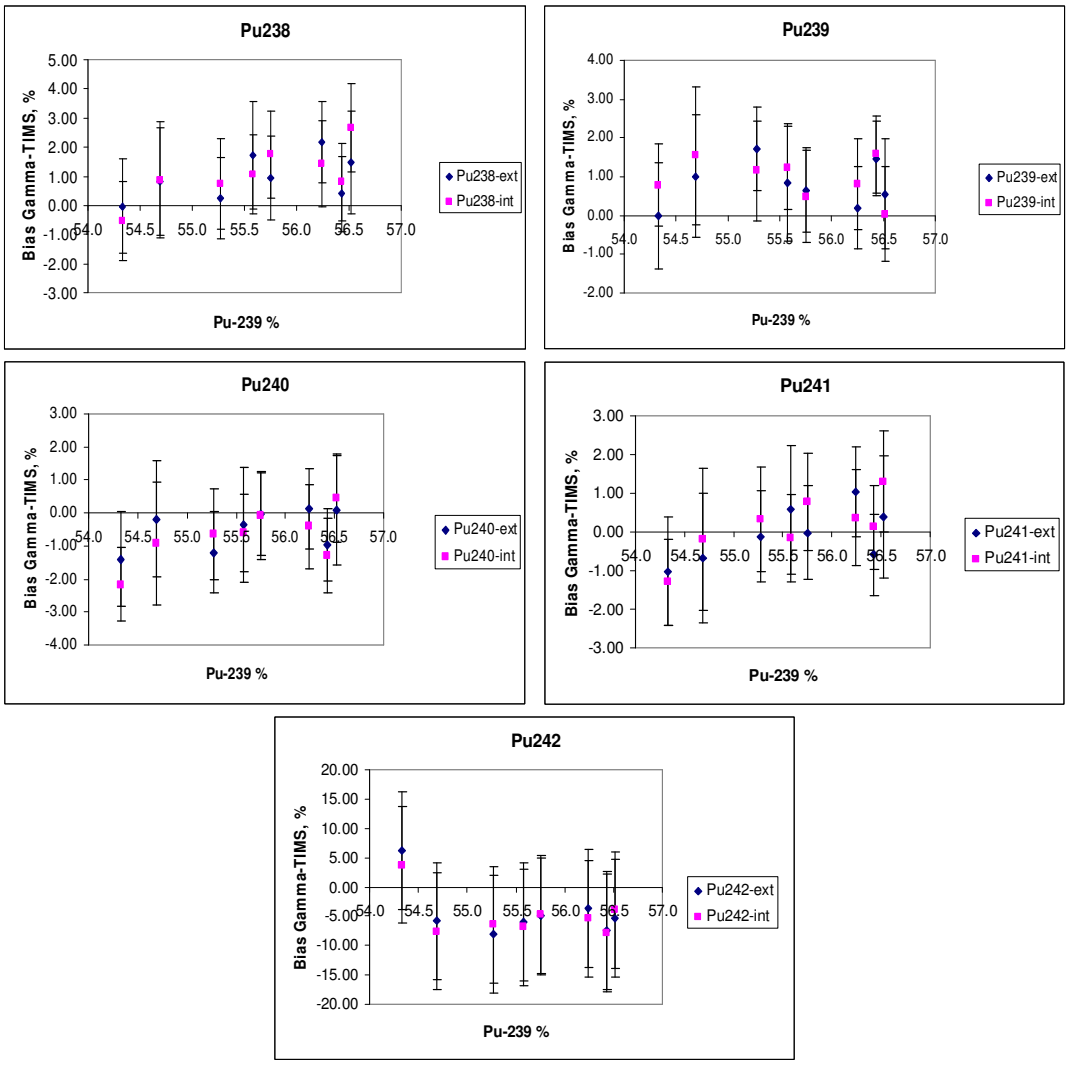


**Figure 1** The relative difference of the Pu isotopic composition between gamma spectrometry and TIMS as a function of $^{239}$Pu isotopic fraction.

**Table 1** Pu isotopic fractions average bias between gamma spectrometry and TIMS for the two gamma spectrometers

| | Pu 238 | | Pu 239 | | Pu 240 | | Pu 241 | | Pu 242 | |
|---|---|---|---|---|---|---|---|---|---|---|
| | Avg. bias | St. dev. | Avg. bias | St. dev. | Avg. bias | St. dev. | Avg. bias | St. dev. | Avg. bias | St. dev. |
| "External" | 0.97 | 0.77 | 0.81 | 0.59 | -0.49 | 0.61 | -0.06 | 0.70 | -4.39 | 4.49 |
| "Internal" | 1.09 | 0.92 | 0.95 | 0.53 | -0.71 | 0.79 | 0.15 | 0.77 | -4.88 | 3.75 |

## 3. Determining the Pu concentration

### 3.1 Calorimetry

Calorimetry provides a very precise non-destructive option for measuring the Pu content of small solid samples (powders and pellets), provided that no chemical reactions are going on in the samples. The decay heat of these samples can be measured, in general, to a precision better than 0.4-0.5%. The decay heat is directly proportional to the mass of the decaying isotopes. Different Pu isotopes have different decay heat; therefore, one needs to know the isotopic composition of Pu in order to convert the measured heat to Pu mass. In addition, $^{241}$Am, which is a decay product of $^{241}$Pu, has a very high specific decay heat, so one needs to know also the $^{241}$Am/Pu ratio in order to calculate how much of the measured heat is coming from Pu and how much from $^{241}$Am. If accurate information on the Pu isotopic composition is available (e.g. from TIMS and alpha spectrometry), then the Pu mass can be evaluated from the measured decay heat to an accuracy better than 1 %. If only isotopics from gamma spectrometry are available, the accuracy of the Pu mass measurement by calorimetry is around 1-2 %.

The eight MOX samples were received each in its own brass transport container all packed together in a single double bag. In our laboratory the whole package was introduced into a glove-box, where the samples were separated, but they were kept in their original transport containers.

As mentioned above, calorimetry provides the Pu mass of the samples. In order to determine the Pu concentration from the calorimetric measurements, on needs to measure the total sample mass as well. Weighing the samples is crucial in this process, as the weighing uncertainties directly translate to the uncertainty of the Pu concentration.

The weight of each pellet has been measured on an analytical balance. For weight measurement the pellets were taken out from their transport containers, inside the glove-box. The containers of the powder samples were not opened during the NDA measurements, therefore their declared weight was taken for calculating the Pu concentration.

While the pellets were out of their transport containers, they were subjected to neutron coincidence counting measurements (see Section 3.2), after which they were placed back to their transport containers

For the calorimetric measurements the MOX samples were kept in their original transport containers. As the calorimeter available in our laboratory is currently not attached to a glove-box, the samples had to be bagged-out from the glove-box for the calorimetric measurements. They were bagged-out inside their containers one-by-one, in very small double bags. The double-bags needed to be very small in order that the samples could fit into the measurement cell of the calorimeter. We used a TAM III calorimeter from TA Instruments. In order to reach thermal equilibrium, the samples were left in the calorimeter for 2-3 days.

### 3.2 Neutron coincidence counting (NCC)

The neutron-gamma counter (”OSL counter”) already mentioned in the section related to gamma spectrometry, was also used for measuring the total and time-correlated neutron radiation of the samples, by means of a neutron coincidence counting electronics. The rate of time correlated neutrons is proportional to the effective $^{240}$Pu mass. To convert $^{240}Pu_{eff}$ to total Pu mass, one needs to know the Pu isotopic composition, just as with calorimetry.

Furthermore, to relate the measured signal to the effective $^{240}$Pu mass, the neutron counter has to be calibrated. For calibration one could use standards similar to the measured samples. However, due to the great variety of samples, with varying isotopic composition, geometrical form, matrix composition etc. this approach can become impractical, as a very large number of calibration standards would be needed. An alternative, followed in the present work, is to use Monte Carlo modelling to calculate correction factors which relate the calibration done with a physical standard to a calibration applicable to the configuration of the measured samples [5].

### 3.3 Isotope dilution mass spectrometry (IDMS)

The first destructive technique used for the quantification of U and Pu concentration was Isotope Dilution Mass Spectrometry (IDMS) technique. Adding to the unknown sample a solution containing the element to be investigated in a validated concentration and isotopic composition (spike) allows a very precise determination of the concentration of the investigated element. The isotopic composition of the spike is chosen as different as possible from the expected isotopic composition of the sample.

The MOX samples were spiked with $^{233}U$ and $^{242}Pu$. Accurate weighing in the spiking procedure is a requisite for precise results in IDMS, so it has to be carried out to the best practices. The so-called "double weighing reverse pipetting method", was our procedure of choice due to its proven higher precision over other methods.

After the addition of the spikes' aliquots to the sample aliquots, a column separation was carried out to collect the U and Pu fractions, which were subsequently analysed by thermal ionization mass spectrometry and, for the Pu fractions, by alpha spectrometry. In fact, as $^{238}U$ is usually present in these samples by a factor of 100 to even 1000 higher than Pu, even a small percentage of contamination can generate a big background at the predominant m/z ratio 238 in the mass spectrometer. The chromatographic separation used cannot guarantee a 100 % separation of plutonium from uranium. In order to solve the problem of the isobaric interference of $^{238}Pu$ with $^{238}U$ in the mass spectrometer, the $^{238}Pu$ to $^{239/240}Pu$ ratio is measured by alpha spectrometry.

The separation was carried out with UTEVA (®Eichrom) resin, conditioned in 6M $HNO_3$. Before separation, $H_2O_2$ was added to the samples, so that Pu was converted into its tetravalent state. For the reaction to be effective, at least 45 minutes are required. Then 1 mL of sample is loaded on the column. 6M $HNO_3$ is used to wash the columns after loading, to remove fission products and Am. Subsequently, the Pu fraction is stripped from the column with a solution of Hydroxylamine and Ascorbic acid in 2M $HNO_3$ and collected for analysis.

Before collecting the U fractions, the columns are washed with 6mL of the Pu stripping solution, to reduce as much as possible the presence of Pu leftovers in the U fraction. U is then stripped from the columns by means of an ammonium oxalate solution and is ready for MS analysis.

### 3.4 Titration

In addition to IDMS, the other destructive technique which we used for determining U and Pu elemental concentration was titration.

The Uranium content of the MOX samples was determined using potentiometric indirect titration following a modified Davies and Gray methodology. The samples were dissolved in 8M nitric acid and later adjusted to reach concentrations around 40 mg U/g solution suitable for titration. The determination of the U mass fraction in MOX samples is possible for a U/Pu ratio higher than 2.5. The plutonium content in the dissolved samples, having at least 10 mg Pu/g solution, was determined using potentiometric back titration. A minimum amount of 0.5 g of solid sample or 10 ml of liquid sample ($HNO_3$ solution) are required.

## 3. Results

The relative differences of the Pu concentration obtained by the applied techniques with respect to IDMS result are presented in Table 2 and in Figure 2. The NDA option presented here is the fully non-destructive option, i.e. the combination of calorimetry with gamma spectrometry. The calibration of the neutron counter by Monte Carlo modelling is in progress and will be presented elsewhere.

One of the powder samples (sample id. 17891) contained almost 2 g of Pu so the heat output of the sample was quite high and it was on the edge of the usual range of applicability of the calorimeter available in ITU. Consequently, the bias for this sample is larger than for the other samples, as it can be seen from Table 2 and Figure 2. Therefore, repeated measurements with smaller, precisely

weighed amounts of powder were done. The measurements on the subsamples provided much better agreement with the IDMS results.

**Table 2** Relative differences of the Pu concentration obtained by various techniques with respect to IDMS.

| Sample Id | Sample form | Calorimetry + isotopics from gamma | | Calorimetry + declared isotopics | | Titration | | Declared | |
|---|---|---|---|---|---|---|---|---|---|
| | | Value | Abs. Unc. | Value | Abs. Unc. | Value | Abs. Unc. | Value | Abs. Unc. |
| 17915 | powder | 0.86 | 0.85 | 0.42 | 0.98 | -0.33 | 0.16 | -0.16 | 0.19 |
| 17921 | pellet | -1.27 | 0.91 | -0.50 | 0.98 | -0.11 | 0.16 | -0.11 | 0.11 |
| 17933 | pellet | -0.55 | 1.20 | -0.14 | 1.00 | 0.16 | 0.21 | -1.79 | 0.13 |
| 17891 | powder | 1.40 | 1.28 | 0.80 | 1.01 | 0.23 | 0.16 | 1.83 | 0.09 |
| 17927 | powder | 1.16 | 1.13 | 0.15 | 1.05 | 0.25 | 0.16 | 0.85 | 0.17 |
| 17903 | pellet | -0.46 | 0.94 | -0.68 | 1.00 | 0.09 | 0.16 | -0.74 | 0.13 |
| 17897 | pellet | -0.15 | 0.93 | -0.49 | 0.96 | 0.06 | 0.16 | 0.32 | 0.18 |
| 17909 | pellet | -0.87 | 1.26 | 0.03 | 1.00 | 0.33 | 0.16 | -0.35 | 0.12 |
| **Average** | | **0.01** | **1.06** | **-0.05** | **1.00** | **0.09** | **0.17** | **-0.02** | **0.14** |
| **Average of abs. values** | | **0.84** | | **0.40** | | **0.20** | | **0.77** | |
| **Standard deviation** | | **0.99** | | **0.51** | | **0.22** | | **1.07** | |

Combining calorimetry with the isotopic composition declared by the operator gives a result which is even closer to the reference values from IDMS, than the combination with gamma spectrometry. Since the consistency of the operator-declared isotopic composition can be verified by gamma spectrometry, the combination of calorimetry with the operator-declared isotopic composition could be also a viable option for the safeguards verification of the Pu content of MOX fuel.

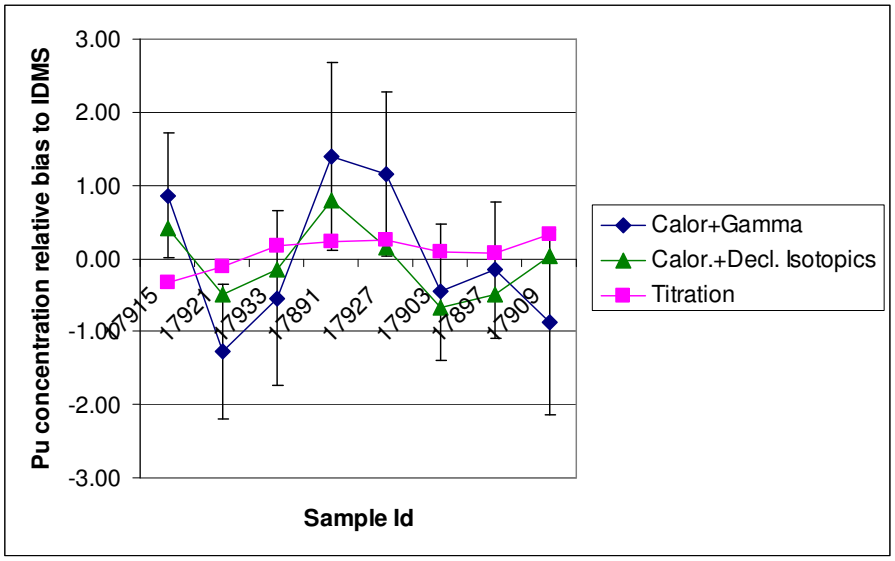


**Figure 2** Relative differences of the Pu concentration obtained by various techniques with respect to IDMS. The error bars (1s) for the combination of calorimetry and gamma spectrometry only are shown.

We remark that the combination of calorimetry with isotopic composition determined in our lab by TIMS and alpha spectrometry (not shown in Figure 2 and Table 2) demonstrates worse agreement with IDMS than the combination with operator-declared isotopics. The reason for this is a negative bias of about 1.2 % which we observed in our alpha-spectrometric measurement of the $^{238}$Pu to $^{239/240}$Pu ratio. This translates directly to a bias in $^{238}$Pu abundance, which has a strong impact on the accuracy of the Pu mass calculated from the calorimetric measurements. The bias in the alpha-spectrometric measurements was due to relatively bad resolution of the spectrometer and

inappropriate peak area evaluation. In measurements of future samples this bias will be eliminated through the use of a better spectrometer and better peak fitting.

## 4. Conclusion

The results in Table 2 and Figure 2 show that calorimetry combined with isotopic composition from gamma spectrometry provides a fairly accurate measurement option for Pu concentration, with an accuracy of about 1%, in a fully non-destructive way.

It is interesting to note that, on average, the non-destructive measurement options show a better agreement with IDMS, than the declared concentrations.

The planned step forward is to re-evaluate all the above described measurements based on the so-called "CANEGA" approach [8], which combines three non-destructive techniques, to simultaneously determine the Pu mass and the Pu isotopic composition, including $^{242}$Pu abundance, without any prior knowledge of the type of Pu under assay. The term "CANEGA" stands for CAlorimetry, NEeutron coincidence counting and GAmma spectrometry. The combination of these three techniques could improve the accuracy of the non-destructive measurement of Pu isotopic composition, and consequently of the accuracy of the non-destructive measurement of the Pu concentration [8].

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# UK-Norway Initiative (UKNI) approach for the development of a Gamma Ray Attribute Measurement System with an integrated Information Barrier

Keir Allen[1], Steinar Backe[2], David M Chambers[1], Edward Day[1], Styrkaar Hustveit[3], Kjell Johansson[2], Colin Marsh[1], Tore Ramsøy[2], Doug Reynolds[1], Aled Richings[1], Hans-Erling Torkildsen[4],

[1] AWE, Aldermaston, Reading, Berkshire, RG7 4PR, United Kingdom
[2] Institute for Energy Technology, P.O. Box 40, N-2027, Kjeller, Norway,
[3] Norwegian Radiation Protection Authority, P.O Box 55, N-1332, Østerås, Norway
[4] Norwegian Defence Research Establishment, P.O. Box 25, N-2027, Kjeller, Norway

***Abstract***

*Any future nuclear weapon disarmament agreement will need to be underpinned by a verification regime that can demonstrate, with confidence, that nuclear disarmament has taken place. Researchers from the UK and Norway have been collaborating, since 2007, to understand the issues involved in the development of 'mutually trusted' instrumentation to perform verification measurements. One aspect of the collaboration has been to develop an information barrier to be deployed as part of a gamma radiation measurement system. This measurement system determines the presence of plutonium and compares the isotopic composition against a declared threshold, whilst preventing the release of sensitive or proliferative information. The achievement of mutual trust requires that both parties clearly agree upon the operation of the measurement system and function(s) of the associated information barrier (IB). The agreed functionality should be implemented in a mutually acceptable manner. It is necessary that additional functional capability be eliminated from the system and that all parties can ensure that this is the case with a high degree of confidence. The system must also be deployed in a manner that does not compromise its intended function. This paper presents the approach taken by the UKNI project team in the development of the measurement system to date.*

**Keywords:** UKNI, information barrier, authentication, joint design

## 1. Introduction

The UK-Norway Initiative is an ongoing collaboration between a Nuclear Weapon State (NWS) and a Non-Nuclear Weapon State (NNWS) which seeks to investigate technical and procedural challenges associated with a possible future nuclear warhead dismantlement verification regime [1]. The collaboration examines how a NNWS might engage with the verification process.

One such challenge is how to overcome the tension created by the opposing requirements of the parties involved. Those charged with monitoring the successful implementation of a regime (here on referred to as 'monitors' or the 'monitoring party') will require a sufficient amount of information and data that demonstrates the other party's adherence to the regime terms. The objective of the monitoring party is therefore to ensure that a sufficient amount of accurate and truthful data has been obtained so that it can be assessed for compliance.

Conversely, the party under inspection (here on referred to as the 'host' or 'host party') must maintain their security arrangements surrounding warheads which may require 'sensitive' information to be withheld from disclosure Additionally, both parties may need to consider their non-proliferation obligations as set out in articles I and II of the Non-proliferation Treaty [2]. Both parties may require

proliferative information to be withheld from disclosure as a result. The aim of the host party is to protect any sensitive information and data whilst complying with the stipulations of the agreement.

The underlying conflict is how the disparate objectives of each of parties may be met, when instrumentation intended to support the monitoring process, is deployed in a 'sensitive' host facility. It is evident that neither party may be content with the *assumption* that monitoring equipment could not be manipulated to undermine each of their objectives.

The monitoring party has the requirement to establish confidence that data collected or results presented are accurate and genuine. They must be able to ascertain that monitoring equipment performs as expected and has not been tampered with by the host party. Ensuring this may entail a process of design verification and engineering tests. The monitor party must also be able to maintain confidence in the fidelity of the equipment for the extent of the monitoring period, This may be have to be accomplished by implementing and maintaining a robust 'chain of custody' over equipment throughout its lifetime.

Concurrently, the host party must be satisfied that use of the instrumentation cannot lead to the disclosure of 'sensitive' information that has not explicitly been agreed upon between the parties for inclusion in the monitoring process. The host party will also be particularly concerned with ensuring the safety of their personnel and assets, and will ensure items introduced into the facility have not been deliberately modified or altered to cause harm.

For instruments or equipment made available by the monitoring party, the host may require a period of intrusive (and private) investigation, including their own design verification and engineering tests. Such testing could destroy any confidence monitors may have previously established in the instrument. Moreover, investigations may be implemented on any and all individual instruments destined for 'sensitive' facilities. The investigation process is also required to ensure the instrument meets facility standards and requirements, though engineering validation may not need to take place in private.

Equipment that passes the investigation of the host party may receive 'Certification' which authorises the use of that individual item within designated facilities under specified operating conditions or procedures. Certification relates to all facility concerns, including but not limited to health and safety, security and engineering validation. If the host party has reason to believe the item has been tampered with, or that its status has changed since the initial investigation, then the certification of the item (or items) may be withdrawn. A deviation from specified operating procedures could be considered sufficient reason to warrant the removal of an item from the list of certified items. Clearly this is a contentious situation that could lead to an impasse between the parties if it has not been anticipated and solutions proposed beforehand.

The UKNI has undertaken to improve understanding of the involvement required by both parties, during each step of the design and deployment of inspection instrumentation, in order to meet each of their objectives. The project has been investigating this through the development of a gamma radiation measurement system incorporating an 'information barrier', which may be relevant to the verification of nuclear warheads in future.

## 2. Gamma radiation attributes and use of information barriers

Given the nature of nuclear weapons, it is evident that certain features of a weapon might be considered sensitive. A number of 'attributes' could be chosen which sufficiently distinguish treaty accountable items (e.g. warheads) from other objects. The host party could declare a set of attributes that provide confidence to the monitoring party that the item under consideration is indeed a treaty accountable item. A method of allowing the monitoring party to perform measurements on treaty account items could then be implemented; to ensure presented items are consistent with the declaration. The measurement on treaty accountable items might lead to the release of sensitive information if free access is allowed to the data produced. This requires that a method is devised for preventing the release of sensitive information whilst enabling attributes to be measured.

An information barrier (IB) is a system, designed to prevent sensitive information from being disclosed under circumstances such as those described [3,4,5,6,7]. An IB may consist of both technical and procedural elements, used to protect sensitive information from disclosure whilst also enabling monitors to confirm agreed attributes.

Fissile materials produce characteristic gamma ray radiation signatures that can be used to confirm the presence and isotopic make-up of the fissile material. These attributes may be relevant to nuclear warheads because fissile material is a characteristic of a nuclear warhead. A possible complication exists because a high-resolution gamma spectrum of a nuclear warhead could contain additional, 'sensitive' information. Consequentially, methods of extracting the declared attributes from the raw data collected are warranted.

The exact isotopic composition of the fissile material may be regarded 'sensitive' by the host party. This being the case, it might be suitable to confirm that the material composition contains a ratio of isotopes that can be compared to an agreed upon threshold. The threshold ratio is arbitrary for research purposes, and therefore the project concentrated on achieving a sufficiently accurate and trusted measurement of an isotopic ratio on non-sensitive material.

It should be noted that, in isolation, confirmation of the presence of plutonium and that the isotopic ratio exceeds (or is a value less than) a given threshold, may not be sufficient to confirm an item as a treaty relevant warhead. Nonetheless warheads might be distinguished sufficiently when these two attributes are combined with other attribute measurements, so measuring material presence and confirming an isotopic ratio may form part of a suite of necessary attribute measurements.

## 3. Building a 'trusted' IB system

Previous studies, established to develop IB systems that measure these attributes (and more), have been documented and discuss features that could help with building 'trust' [4,5,6,8,9]. In 2007 the UKNI undertook to develop an IB system that could determine the presence of plutonium and confirm whether an agreed isotopic ratio had been reached, in a manner that allows both host and inspector (and, potentially, third parties) to place a high level of confidence in the validity and veracity of any result obtained. The aim has been to produce a system in which the host can be convinced that no information could be communicated other than that explicitly agreed; the monitors that the answer given is indeed 'truthful' and suitably accurate. Therefore, the objective of the project has been to develop a more detailed understanding of how to implement a process that builds a suitable level of mutual confidence in the development of a measurement system.

## 4. Initial requirements

The project was initiated in 2007 with a number of key requirements that have impacted upon the direction taken:

1. The design of the measurement system incorporating the IB must be a joint design:

   The underlying assumption was that full design knowledge would allow both parties to establish a baseline expectation about the capabilities and limitations of system; from which the parties could establish and maintain confidence. Mutually acceptable standards and procedures for the construction, testing and operation of the instrument could be established. In addition, assurance that the design meets all necessary safety and engineering requirements for the facility for which it is intended could be provided, because the host party would have an equal voice in the design specification.

2. 'Simplicity' is desirable at all times:

This requirement was based upon the assumption that a ‘simple as possible’ design would be ‘easier’ to understand. The potential to identify subversive deviations from the jointly agreed design may be higher for a ‘simple’ system. By limiting the instrument complexity, design authentication, and engineering verification and validation, might be more easily accomplished in a mutually acceptable manner.

The stipulation of these two requirements was made to help both parties establish confidence in the system. An addition third key requirement was specified:

3. Confidence must be built upon ‘host supplied’ equipment:

Given the certification process outlined in the introduction, the assumption was made that any instruments would be ‘host supplied’. This was taken to mean that the first contact of the monitoring party with IB instrumentation or modular components occurs after being presented with products that had been manufactured by the host party to the jointly agreed design specification. Any prior confidence the monitoring party might have possessed in instruments they themselves presented for certification could be destroyed during the certification process, requiring the monitors to start over with building confidence. This could be deemed equivalent to ‘host supply’.

## 5. Development of the IB project

The project has evolved across three phases. At each stage, the joint project team has worked together to produce a mutually acceptable solution to the challenges faced. All three phases of instrumental development have used gamma ray data obtained from a commercially available High Purity Germanium (HPGe) detector.

### 5.1. The phase I system

The phase I system design was completed in 2009 [7]. The purpose of this phase was to provide a platform to discuss how to design a ‘simple’ and system for the collection and analysis of gamma ray data. It also allowed more detailed discussion of possible authentication techniques and features that could help with the building of ‘trust’.

The operational processes performed by the IB were agreed upon within phase I and have remained in place through all phases of the IB development. Though the specific analysis routines have been necessarily modified as lessons from earlier phases have been applied to the system design, the operational process has remained. The flowchart in Figure 1 outlines the agreed upon operational process of the measurement system.

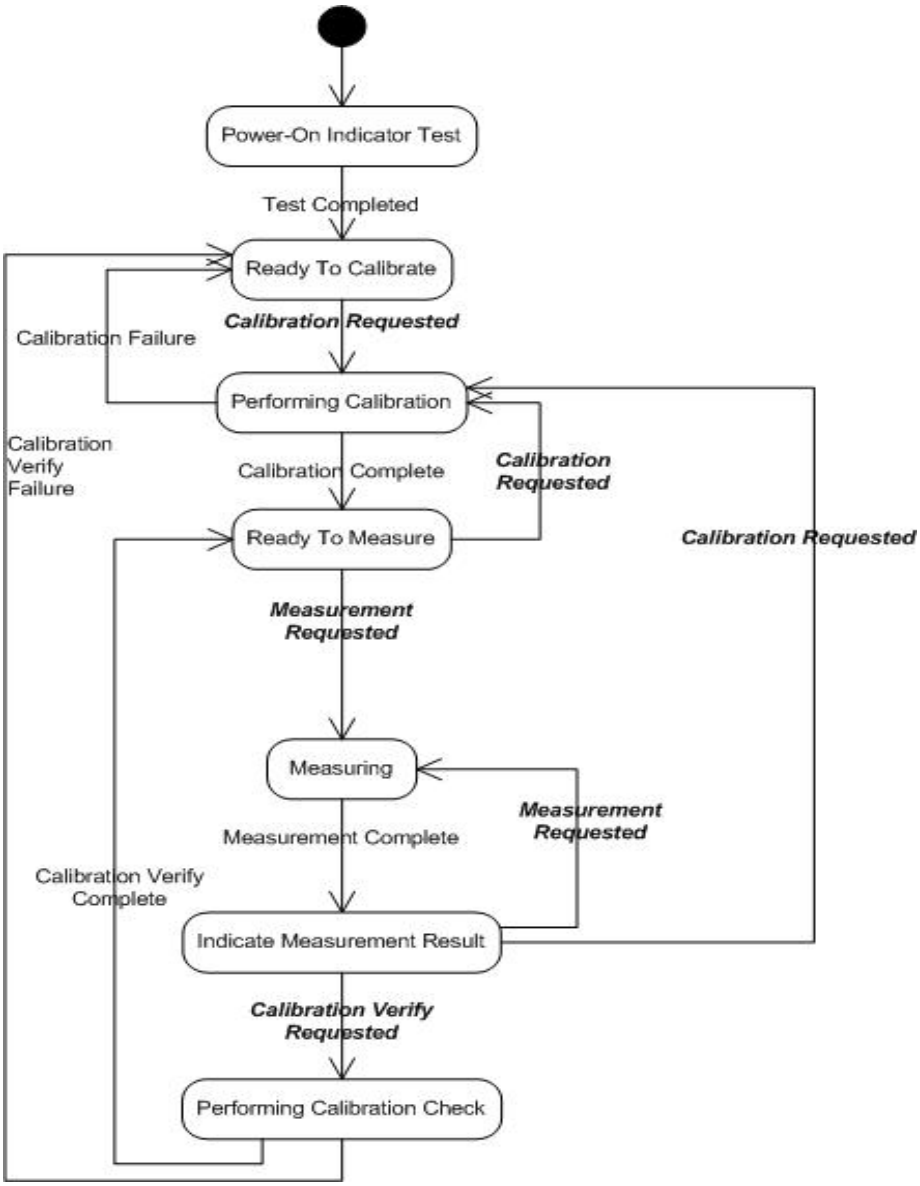


**Figure 1:** Operational Processes of the UKNI Information Barrier

The system was designed to detect the presence of a surrogate radioactive material ($^{60}$Co), by analysing the number of counts detected in the 1173 keV and 1332 keV peaks, above background. The energy range of the instrument was set from 0-1600 keV. A low level discriminator was set at 100 keV to prevent low energy gamma rays from overwhelming the system. The analysis methodology was agreed upon and specified for how the instrument should perform the necessary 'measurement' calculations.

It was necessary to consider how to verify that the specified methodology had been implemented in software correctly. The proposal was to develop two versions of the instrument using different software, in order to investigate alternative approaches to software verification. One version was written in Assembly language, the other in SPARK – a 'safe' subset of Ada [10]. The two languages are still being employed so that verification methods for high level and low level languages can be compared.

Phase I resulted in agreement over components that could be modified for the later phases. Use of an 8-bit microcontroller was agreed upon (since that component possesses a minimal functional capability, in line with the requirement) and the Analogue to Digital Converter (ADC) onboard was used.

### 5.2. The phase II system

Phase II of the system development was initiated in 2009 to investigate how to determine the ratio of the quantity of two materials and to agree a common system design [7].

It became evident that simplifying assumptions adopted in Phase I would not be prudent because these assumptions did not account for the possible presence of intervening materials. The presence of intervening material between the measurement system and the item under consideration presents a challenge to the development of analysis algorithms because of differential attenuation. In a warhead verification scenario, it is evident that unknown materials may be present and this necessitated an additional design requirement to be stipulated:

- The instrument should operate in an environment where undetermined materials and/or thicknesses of material may be present between the gamma source and the HPGe detector.[1]

In order to minimise the differential attenuation it was determined that the energy of the gamma ray lines should be chosen to be within a narrow energy range. Choosing the energy range in this manner also minimises the variation in detection efficiency.

The peaks used during Phase II development were the 1275 keV from the $^{22}$Na isotope and the 1332 keV from $^{60}$Co, and the background was determined from predefined regions. A mutually acceptable calibration routine was devised and implemented using the 122 keV and 1408 keV peaks from a $^{252}$Eu source.[2] The energy range was kept at 1600 keV, but the onboard ADC was disabled and an additional 12 channel ADC was introduced to provide better resolution during phase II, with 0.39 keV per channel.[3]

A high-activity source used during phase I tests overloaded the preamplifier and prompted the inclusion of a count rate monitoring mechanism in phase II, with a dynamic range of 10 - 10,000 counts per second.[4] The necessity for a minimal counting period was also discussed to ensure that the counting statistics for the ratio measurement were reasonable. This was implemented by ensuring counting continued until the number of counts in the 1275 keV peak reached 10000.

## 6. The phase III system

Phase III of the work started in 2010 and has focussed on how to implement a methodology to confirm the presence of plutonium, and determine the isotopic ratio of $^{240}$Pu/$^{239}$Pu. Development within of phase III has progressed markedly slower than phase II, this in large part is due to the difficulty in obtaining suitable plutonium samples and facilities in which to carry out the measurements necessary for system development. This highlighted an additional benefit of using surrogate sources which are widely available in many scientific establishments: Systems and methodologies can be developed and tested more freely.

The 8-bit microcontroller used in previous phases has also been used in the phase III version of the instrument. The limited capability of this hardware has presented some challenges in developing an accurate analysis approach that can be implemented efficiently. Even so, it is possible to discuss the challenges faced and acceptable solutions that could satisfy both host and monitor party requirements for transparency in the development process.

[1] Complete attenuation, though possible, is discounted, since this would in effect equate to the declaration of an attribute that could not be verified.
[2] The same calibration routine was later applied to the system developed under phase III
[3] This energy range and resolution has remained fixed through phase III.
[4] The count rate monitoring mechanism was modified during the phase III, with a count rate of 50 -50,000 counts per second.

## 6.1. Measurements

Specifying how the IB should perform the calculations necessary to determine presence and isotopic ratio has been challenging because of the complexity inherent in real plutonium spectra.

Part of the warhead verification challenge resides in the possibility that no further information will be forthcoming regarding the gamma ray data collected, besides that in the declaration made by the host. Discussions have taken place concerning the potential effects of the presence of undetermined sources. A suitable methodology should only use the information in the declaration; other sources that could potentially contribute to the pulse height distribution must be adequately anticipated and accounted for within the algorithm.

Identifying suitable methods for determining plutonium presence and an isotopic ratio have been challenging given the hardware constraints, but agreement has been reached for each of the functions required. Since phase III development is ongoing, approaches are subject to testing and refinement.

## 6.2. Plutonium identification

It has been necessary to consider which peaks are most suitable for confirming the presence of plutonium, whilst anticipating potential shielding by unknown intervening materials, and also ensuring that a suitable number of counts will be collected in a reasonable counting time. Peaks at low and high energies are less favourable in this situation.

The presence of $^{239}$Pu will be confirmed if peaks are present at the energies of some of its more intense characteristic gamma ray lines. Determining with confidence that plutonium is the source of these peaks is relatively easy for a gamma spectrometry specialist, but implementing suitable automated tests that provide confidence, within the hardware constraints, has proven to be a challenge, and work continues in this area. It has not been determined whether sources other than $^{239}$Pu *could* replicate those peaks sufficiently to pose as an effective spoof.

## 6.3. Isotopic ratio measurements

An aim has been to design a system that could work for any given plutonium isotopic ratio. This is to ensure the system would be applicable for any future measurement where the actual ratio would be declared by the host party. A challenge has been how to determine the $^{240}$Pu/$^{239}$Pu ratio with sufficient fidelity so that the results are useful, given the restrictions of the hardware chosen for the project.

A region of the plutonium spectrum is required that contains peaks from both $^{239}$Pu and $^{240}$Pu, which are likely to be detected through a quantity of shielding and contain sufficiently high count rates.

The 642.35 keV line is the only suitable $^{240}$Pu line that meets these two criteria. The $^{239}$Pu line chosen is the 645.89 keV line, and the methodology compares the two. The two peaks are located sufficiently close together on the energy spectrum to simplify the problem posed by the unknown shielding attenuation coefficient. The 620 – 670 keV region of a plutonium pulse height distribution containing these two energy peaks is pictured below in figure 2.

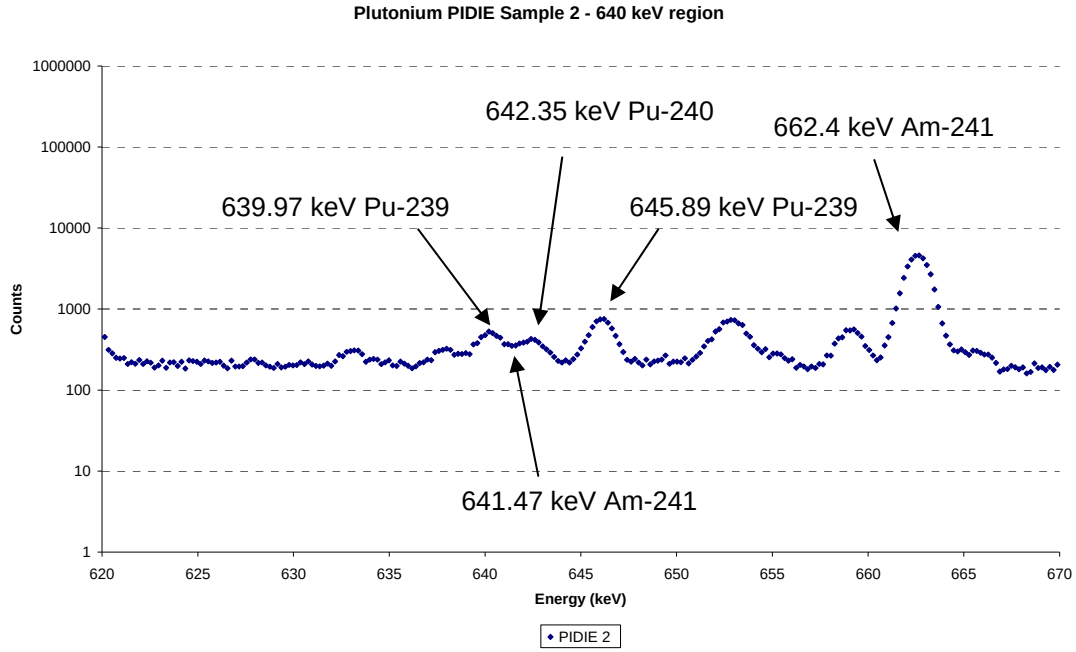


**Figure 2:** A pulse height distribution obtained from a nonsensitve Pu source used to illustrate the plutonium 640 keV region. The pulse height distribution was colected from one of the Plutonium Isotopic Determination Inter-comparison Exercise (PIDIE) standards.

A comparison between the energy lines is complicated because of the energy resolution of the detector used, which leads to peaks that overlap in the pulse height distribution. Counts from the $^{239}$Pu 639.97 keV line contributes to the peak area of the $^{240}$Pu 642.35 keV peak in the pulse height distribution. An energy line is also present at 641.47 keV due to $^{241}$Am, which also contributes counts to the 642.35 keV peak. A direct comparison between the counts recorded is therefore unsuitable and a method of determining the contribution of all three energy peaks to the counts recorded is required if an accurate determination of the ratio is to be made.

Gamma Spectrometry tools more traditionally analyse this region using complex peak fitting algorithms which are beyond the capability of the IB system. The approach chosen to determine the ratio does not use computationally extensive peak fitting routines to determine the nature of the item under consideration. The number of counts detected in $^{241}$Am 662.4 keV peak and in the $^{239}$Pu 645.89 keV peak, along with the branching ratios for americium and plutonium, are used to determine the contribution of each to the $^{240}$Pu peak at 642.35 keV. The contributions from the $^{241}$Am and $^{239}$Pu in the 640 keV region of interest (635.16 to 643.75 keV) are subtracted out to determine the $^{240}$Pu contribution.

The determination of the isotope ratio be the IB must be accurate if it to be meaningful in a measurement. In order to ensure a minimal number of false negatives, an inaccurate system could require the threshold ratio to be set where even low quality material could pass the test, rendering the system useless. The uncertainties associated with the algorithm contribute to the rate of false positive and false negative results and so should fall within a suitable, agreed limit. Work is ongoing to determine the level of uncertainty associated with the algorithm and the accuracy of the system, and will be reported in due course.

## 6.4. Background subtraction

The task set during development of phases I and II was simplified by specifying regions that defined the background. For phase III an additional requirement has been determined: That the background count rate should be determined as part of the measurement process.

This has been stipulated because it cannot be assumed that the treaty account item would be absent during a separate measurement of the background count rate. Also, it may be difficult to ascertain that any separate background measurement is a true representation of the background during the attribute measurement. An interpolation technique has been implemented to subtract the underlying signal from the peaks.

### 6.5. Counting statistics

The algorithm has been set to run for an agreed amount of time and report at the end of that defined period, to prevent any additional communication concerning the strength of the gamma ray source.

One of the considerations has been how to ensure reasonable counting statistics for each of the functions performed by the IB whilst maintaining a total number of counts below the limit imposed by the hardware memory. The solution which has been adopted was to split the two functions performed by the IB. Since these functions use different regions of interest (ROI) in plutonium gamma spectra, splitting the operation maximises the counts collected for each function under the hardware constraint.

The identification function uses a key peak found due to the 413 keV line of $^{239}$Pu. Because of the intensity of this line it is likely that the peak found at this energy in the pulse height distribution will contain the most counts, compared to other peaks characteristic of Plutonium. The IB hardware can process a maximum number of counts in any one peak and the 413 keV peak is considered the most likely to reach the maximum number. In recognition of this, a limit has been placed on the number of counts attained in that peak. If this limit is reached during the measurement process then no more additional counts are recorded in any of the 'identification' regions of interest.

Data continues to be collected in the 'isotopic' region of interest until the timeout period has been reached. The branching ratios for the Pu lines in the 620 – 670 keV region are orders of magnitude less than for the 413 keV line, so memory capacity should not be reached.

### 6.6. Algorithm implementation (verification of software)

With the specification agreed for how the IB should perform each calculation for each operation, a method for sufficiently demonstrating that this specification has been accurately and faithfully implemented in software is required. This will allow confidence to be built in the correct functioning of the IB, but this element of building mutual confidence might be very challenging in any case.

The software development for the project has followed distinct paths so that different approaches could be tested. Individual developers have been given considerable flexibility over the style of implementation. This has been beneficial in the research environment since it allows for different approaches to be tested.

Nevertheless, providing that the (jointly agreed) specification is rigorous and complete, correct implementation should be demonstrable, which will benefit the authentication process. Development of software verification measures is underway, based upon mathematically rigorous analysis tools, supported by procedural selection approaches [11]. Details of this work will be reported in due course.

## 7. Future work

There is still considerable work for the UKNI IB project team to undertake before a full report can be made available. Analysis of trials data is necessary to understand the performance of the instrument against plutonium. Following on from this, any improvements that can be implemented within the algorithm will need to be investigated. Development of a mutually acceptable approach to authenticating the IB software is continuing.

Both parties must be capable of maintaining sufficient confidence in the status and operation of the IB at all relevant points during its lifetime. This requires the system to be deployed in a manner that does not compromise its intended function. Both parties must be satisfied that' at no point in its lifecycle' could an instrument be used to either convey incorrect or manipulated results, or collect or convey more information than had been explicitly agreed for that instrument. The team is also working on a set of procedures to cover the lifetime of the instrument that will allow both parties to maintain confidence once it has been established.

# 8. Discussion and conclusions

## 8.1. Balancing objectives

Due to the their respective objectives concerning collection and protection of information, neither host or monitoring party are likely to inherently trust equipment to be deployed during a verification process, or assume the instruments operate sufficiently, accurately and truly without verifying this to be true. This is a typical approach found in safeguards and arms control: 'trust but verify'. In a warhead dismantlement context the host may insist on supplying all equipment to help counter their concerns. The situation as described may require a method of developing confidence in the systems deployed; confidence which *both* parties must be able to develop and maintain. The UK-Norway Initiative has been exploring this problem through the development of a gamma radiation attribute measurement system that incorporates a bespoke 'Information Barrier'.

The task an information barrier performs will be specific to the question asked and unique to the circumstances in which it is required. An information barrier will form part of a bespoke system. It is not simply a physical instrument that can be applied to any given situation since the preferred design solution under one regime may not adequately address the specific question or concerns relating to a regime with different starting conditions.

Nevertheless, development of bespoke equipment with specific requirements is a good method for understanding how to achieve 'mutual confidence'. In order to build a mutually acceptable system, the parties involved must acknowledge each others objectives and should all possess a sufficient level of knowledge concerning the construction and function of the IB to ensure their own objectives are consistently met. Ensuring this can become much more complex as the functional requirement of the instrument increases.

## 8.2. The UKNI information barrier

The UNKI has tried to build mutual confidence in the functioning of a system through transparently designing and building a system. The capacity of the IB to perform subversive or unauthorised actions is limited, in part, by limiting the processing capacity of the hardware.

The choice of hardware has limited the capability of the instrument in line with initial requirements, perhaps at the expense of implementing suitably capable measurement functions. A particular challenge for the project team has been to measure the $^{240}Pu/^{239}Pu$ ratio sufficiently accurately, without implementing any further assumptions or inputs; and yet in a sufficiently simple manner, commensurate with the capacity of the hardware to perform the calculation. Future analyses will help with determine the technical capability of the instrument as it has been developed to date.

Importantly, should the current algorithm require improvements, this method of jointly researching and designing equipment can be applied to make any necessary improvements in a mutually agreeable manner. Furthermore, the lessons learnt from this particular project can be applied to many other technologies that may be useful during warhead dismantlement verification. Each party may wish to build confidence in the functioning of such technologies. Other attributes may be important, which may require suitable information barriers to enable other them to be interrogated. Containment and

surveillance technologies that could help maintain the chain of custody over items and equipment could also benefit from the approach.

Progress during phase III has been made much slower due to the need to source plutonium samples. This slow progress is in marked contrast to the rapid implementation made during phase two, made possible by using surrogate materials. Many of the challenges and issues faced to date could be investigated equally well without using plutonium and suggests that rapid progress could be made more widely through research using alternative sources.

### 8.3. Joint development

The challenges in relation to developing technologies in a mutually agreeable manner have been usefully for explored through development of the UKNI IB. It has provided an insight into the resources, expertise and timescales that may be required to produce equipment in a reasonable amount of time following this approach.

In a treaty context, both parties will wish to understand the trade offs and consequences involved in developing a design specification for any instrument. The instrument must be suitable for the operational environment into which it will be deployed, with the design anticipating constraints that may be imposed by that environment. For the system to be effective, both parties must clearly agree upon the functions the instrument performs. The intended functions also have to be suitably implemented within the technology. Additional functional capability should be eliminated and either party should be able to ensure this to a high level of confidence.

The project has led to better understanding of the extent to which parties must work together to accommodate each Party's concerns and build a mutually acceptable system in which both parties can develop confidence.

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# Societal Verification: Intellectual Game or International Game-Changer

**Kelsey Hartigan, Corey Hinderstein**

Nuclear Threat Initiative
1747 Pennsylvania Avenue, NW
Washington, DC 20009
E-mail: hartigan@nti.org, hinderstein@nti.org

***Abstract:***

*Within the nuclear nonproliferation and arms control field, there is an increasing appreciation for the potential of open source information technologies to supplement existing verification and compliance regimes. While clearly not a substitute for on-site inspections or national technical means, it may be possible to better leverage information gleaned from commercial satellite imagery, international trade records and the vast amount of data being exchanged online and between publics (including social media) so as to develop a more comprehensive set of tools and practices for monitoring and verifying a state's nuclear activities and helping judge compliance with international obligations.*

*The next generation "toolkit" for monitoring and verifying items, facility operations and activities will likely include a more diverse set of analytical tools and technologies than are currently used internationally. To explore these and other issues, the Nuclear Threat Initiative has launched an effort that examines, in part, the role that emerging technologies and "citizen scientists" might play in future verification regimes. This paper will include an assessment of past proliferation and security "events" and whether emerging tools and technologies would have provided indicators concurrently or in advance of these actions. Such case studies will be instrumental in understanding the reliability of these technologies and practices and in thinking through the requirements of a 21st century verification regime.*

**Keywords:** Verification, social media, open-source information, arms control, disarmament

## 1. The "Datafication" of Society

The ability to communicate and exchange information globally has changed the way publics interact, businesses operate and governments function. Private citizens can now access high resolution commercial satellite imagery, small business owners can market and ship to consumers all over the world and publics can demand more of their government with a mere 140 characters. Facebook has now exceeded 1 billion users, and every minute, over 100,000 tweets are sent and more than 3,000 photos are uploaded on Flickr.[i]

By tapping into these diverse sources of information, governments and engaged publics are leveraging "big data" to open up new areas of analysis. As the Economist's Kenneth Cukier and Oxford's Viktor Mayer-Schoenberger explain:

> Today, less than two percent of all stored information is nondigital. Given this massive scale, it is tempting to understand big data solely in terms of size. But that would be misleading. Big data is also characterized by the ability to render into data many aspects of the world that have never been quantified before; call it "datafication." For example, location has been datafied, first with the invention of longitude and latitude, and more

> recently with GPS satellite systems. Words are treated as data when computers mine centuries' worth of books. Even friendships and "likes" are datafied, via Facebook. This kind of data is being put to incredible new uses with the assistance of inexpensive computer memory, powerful processors, smart algorithms, clever software, and math that borrows from basic statistics. [ii]

This "datafication" of society, combined with novel tools and technologies that sort, analyze and even predict certain outcomes, have powerful implications for how states and publics might address the most pressing international security challenges of the 21st century.

## 1.1 Arms Control and Nonproliferation Applications

Within the nuclear arms control and nonproliferation arena, the greater connectivity of people in previously isolated areas, improvements in data mining and filtering techniques, and open source information technologies have the potential to increase government transparency, empower nontraditional actors to monitor and report illicit acts, and even assist with verifying a state's compliance with international obligations. As the U.S. State Department's Acting Under Secretary for Arms Control and International Security, Rose Gottemoeller, has explained:

> Our new reality is a smaller, increasingly networked world where the average citizen connects to other citizens in cyberspace hundreds of times each day. Today, any event, anywhere on the planet, could be broadcast globally in seconds. That means it is harder to hide things. When it is harder to hide things, it is easier to be caught. The neighborhood gaze is a powerful tool, and it can help us to verify the treaties and agreements we've created.[3]

Nontraditional stakeholders can increase the likelihood that violations of international commitments are detected. To be effective, the first task is to define the detection goals. In some cases, societal data and devices might be directly relevant to treaty limited items. In other instances, these tools might convey information that is only indicative of potential violations or misuse. Pinpointing the detection tasks is a crucial first step, and can include a range of objectives:

- Defining **patterns** (e.g. of behavior, activities, movements of people and things)
- Looking for **shifts** (e.g. changes of behavior, activities, movements either from the norm or from the expected)
- Identifying **outliers** (a single activity, person or pursuit that doesn't match the expected or predicted)
- Filling in **blind spots** (e.g. where are there gaps in knowledge from traditional verification systems)
- Detecting **signals** (e.g. something which may indicate something else but which is not itself a proscribed activity)[4]

In many cases, open source information derived from societally generated sources is already included in national all source intelligence gathering and analysis. This is in its early stages, however, and it is not clear how this addition to NTM is practically integrated into treaty compliance monitoring and verification.

To study these and other issues, in 2012, the Nuclear Threat Initiative launched a Societal Verification Working Group as part of its Verification Pilot Project. NTI assembled a group of ten experts from multidisciplinary backgrounds to systematically review the potential for open source information technologies to supplement existing nuclear verification and compliance regimes.

For the past year and a half, NTI's Societal Verification Working Group has been examining the opportunities and challenges that accompany a greater public role in verification. While the group

is exploring the utility of social media platforms like Twitter and Facebook, the focus extends beyond this to include data gathered through other non-traditional, commercial sources and analysis from nongovernmental organizations, independent scientists and other nonstate actors.

While the working group's findings and recommendations are not yet complete, the group is working to define a framework for how states might integrate various societal contributions within existing and future verification regimes. Given the rapid growth of the technology and data sectors, it is clear that it is no longer a matter of *if* states should leverage such tools and information, but *how* they should do so.

## 2. Future Verification Needs

Today, existing verification tools and systems are largely effective given the tasks at hand. Yet as future agreements become more complex and involve a greater number of parties, future verification regimes will likely require new and enhanced tools for monitoring and verifying compliance.

### 2.1 Verification Methods and Tools

The two primary methods for verifying existing treaties and agreements are National Technical Means (NTM) and cooperative monitoring measures. National Technical Means (NTM) refers to those measures deployed by states parties independently, including reconnaissance satellites, signals intelligence and electronic intelligence. Cooperative monitoring measures include a variety of systems, devices and technologies that parties jointly agree to within the context of a treaty or agreement. For instance, in 1987, the United States and Soviet Union first agreed to allow cooperative on-site inspections to verify the Intermediate Nuclear Forces (INF) Treaty. The treaty included a detailed set of carefully scripted ground rules for on-site inspections. U.S. and Russian experts have since developed and agreed to use portal perimeter monitoring, tags and seals and other cooperative measures to verify compliance. Similarly, IAEA inspectors utilize a variety of tools, including containment and surveillance and environmental monitoring to confirm facility accounting and operating records.

Future verification regimes are likely to become even more rigorous and intrusive as states move to lower numbers of nuclear weapons and parties need the ability to detect and monitor smaller items and quantities of nuclear material. It will no longer be enough to count and monitor delivery vehicles, especially as states begin to address non-strategic and non-deployed systems. Warheads held in containers in storage facilities will likely need to be monitored and accounted for. In fact, future arms control agreements may require the exact number of warheads to be counted, a metric that inspectors have not used in past agreements. It may even be necessary to monitor the cradle to grave lifecycle of individual warheads and materials, particularly as nuclear power programs expand, and additional facilities and material flows need to be safeguarded.

### 2.2 The Growing Gap

While important progress and technical achievements have been made, parties are likely to encounter strains on existing NTM and other cooperative monitoring measures going forward. Given the likely demands of future verification requirements, states will need to take advantage of all possible tools and sources of knowledge about compliance. Societal verification has a role to play. Nontraditional tools and data sources can add to the verification process, and such sources can also level the playing field as states without extensive experience or NTM join future multilateral agreements.

A larger framework for incorporating societal verification mechanisms with existing regimes in the near term, and phasing in more complex tools and resources over the longer term, will be required if states are to leverage such sources efficiently and cooperatively.

## 3. Societal Verification

The societal verification concept is not new. In fact, such a theory was first introduced in the 1950's as "inspection by the people." Seymour Melman and Lewis Bohn were among the early theorist who outlined provisions for such a concept. Agreements were expected to require participating governments to criminalize violations, develop assurances and protection for individuals reporting on their country, and socialize the public to place international concerns over domestic loyalties.[5]

During the 1990's societal verification gained renewed attention, including when Joseph Rotblat picked up many of the same themes Melman, Bohn and others developed early on, such as legal protections for reporting, the need for broad participation, and the "deeply felt moral obligation" to report violations of a treaty to eliminate nuclear weapons.[6] He also suggested that scientists and technologists in relevant industries could act as watchdogs, both of their organizations and of their colleagues. These early concepts outlined many of the fundamental issues that still need to be addressed today in order to integrate new tools and systems into future verification regimes.

Modern day concepts of societal verification can take on many forms. While the "whistleblowing" component is still an area for examination, technical growth in both data sources and data analysis make more advanced uses of societally generated data more interesting. A multitude of online platforms and the spread of smart phones and other information and communication technologies make it possible to collect and analyze massive amounts of publicly generated data. Public-private partnerships can also provide additional insight into information collected by companies about attempted procurements and activities of potential treaty violators. Outside experts also have increasingly sophisticated tools to perform their own analysis and disseminate their findings and recommendations to a broad, international audience which acts as a check on government conclusions. These resources can provide additional windows of insight and provide increased confidence over the long run that a state is living up to its obligations.

### 3.1 Monitoring and Mobilizing Functions

There are two primary forms of societal verification: passive and active. Passive and active "sensing" have unique characteristics, which are particularly important to distinguish as states begin to utilize such analysis and tools and parties build a framework for future use. Connecting the entities that utilize passive and active sensing – governments and private citizens or corporations – will be an important challenge to overcome since such analysis currently takes place in separate silos.

#### 3.1.1 Passive Sensing

Passive forms of societal verification center on data collection and analysis, and serve a monitoring function. Content from social networking sites, blogs, microblogs and photo sharing sites can be harnessed to better understand public perception and responses to evolving situations. For example, experts at the Pacific Northwest National Laboratory recently analyzed a ten percent sample of tweets to determine the social media footprint of India's April 19, 2012 Agni-V long range missile test. The PNNL team was able to isolate a set of tweets (some in Hindi) focused on the test, which provided some insight into the public reaction and regional security implications of the missile firing. Further analysis of such information might be able to identify whether test preparations would have been visible in the days and weeks prior, particularly if combined with other data streams.

Social media analytics can pinpoint the topic, sentiment and mood of selected conversations, posts or chats, and experts are increasingly able to add a predictive component to such analysis.[7] This data can be harvested and integrated with data from other sources, so as to include other spatial and temporal characteristics which may provide additional insight.

There are several challenges for passive sensing. While the sheer quantity of information can allow for less precision overall, it is still difficult to decipher signals from the noise, particularly in the arms control and nonproliferation context if it is unclear what signals one is hoping to detect. On the other hand, the sheer volume of data has the advantage of quickly isolating outliers – those individual messages or pieces of information that are either in conflict with what the vast majority is reporting or stands out as providing a unique piece of data. This helps to dismiss an anomalous source or narrow attention to a smaller set of data points worth deeper investigation. Language, cultural and socioeconomic barriers, timeliness of collection and analysis and legal questions regarding privacy and data ownership are also important issues to consider.

### 3.1.2 Active Sensing

Active forms of societal verification rely on public mobilization and involvement. Rather than simply collecting data that already exists, active sensing requires the tasking of specific actions to individuals or groups who are otherwise generating unspecific information. For instance, DARPA's 2009 "Red Balloon" Challenge and the State Department's follow-on TAG Challenge asked the public to help identify and report certain items (red balloon and jewel thieves, respectively) that were located throughout the U.S., North America and Europe. [8] The runner-ups of a separate, more recent State Department challenge took this concept a step further and proposed creating online treaty relevant games such as a "Where's Waldo" inspired challenge where individuals, using avatars to mask their identity, would be given points when they identify treaty relevant objects; and another game where users would use geocaching and QR codes to accept and complete challenges from treaty experts.[9] Other forms of active sensing could include engaged citizens purchasing personal technologies, such as a smart phone equipped with an accelerometer, which would allow the user's phone to constantly collect data and send alerts when abnormalities are detected.[10]

Crowdsourcing, ubiquitous sensing, and games or challenges can provide an excellent opportunity to tap into the insight of engaged publics, but data verification, legal and political issues and incentives for continued participation complicate such methods. While mobilization structures hold promise, such forms of societal verification also present the most challenges which may in some cases be insurmountable absent careful and rigorous concept development and implementation.

## 3.2 The Role of Outside Experts

Nongovernmental organizations, independent scientists and other nonstate actors can utilize both monitoring and mobilization strategies. Indeed in many instances, outside experts self-mobilize as certain events arise. As access to commercial satellite imagery expands, for instance, nontraditional actors are providing additional insight and expertise by using this imagery to give the public a better understanding of what is happening in certain countries. For example, the Institute for Science and International Security (ISIS) utilizes satellite imagery to analyze nuclear sites and facilities in Iran, Syria, Israel, Pakistan, India and North Korea. Citizen "scientists" have also proven highly effective and accurate in projects coordinated by the National Archives, Galaxy Zoo, and National Geographic, to name a few.

How such analysis is corroborated and integrated with existing information structures and how outside experts are cultivated and connected will be important issues to consider if third party analysis is formally integrated in future frameworks. Alternatively, it might prove valuable to leave

these communities separate from formal verification systems and allow them to act as both a check and balance to state-level analysis and a canary in the coal mine to identify areas requiring further attention by national authorities.

## 4. Framework for Leveraging Societal Data and Tools

When making compliance determinations, states can integrate passive forms of societal verification with National Technical Means at any point, but over the long-term the goal should be to work towards using societal verification tools for cooperative monitoring measures, better integrating outside analysis and assessing the role of active sensing methodologies.

### 4.1 Incorporating Passive Sensing with National Technical Means

As a mere data collection tool, states can integrate passive societal verification mechanisms with existing NTM structures immediately. In some cases, national intelligence systems are already using some of these data sources and developing advanced analytics. How these developments can be assessed with regard to their applicability to treaty verification is an outstanding question.

In 2010, the Nuclear Threat Initiative published *Cultivating Confidence*, a study focused on verification, monitoring and enforcement for a world without nuclear weapons, which also laid the groundwork for NTI's ongoing Verification Pilot Project. One area of focus included on how states might verify the nonproduction and elimination of fissile material for weapons. Here, Annette Schaper emphasized the contribution engaged citizens can make through societal verification. Schaper suggested states incorporate societal verification with National Technical Means since NTM is broadly understood as "collecting any information that could help draw conclusions regarding a treaty partner's compliance." Schaper explains:

> National technical means is a well-accepted additional tool of verification that is carried out by states, initially without visible international cooperation. The term is officially included in many arms control treaties and denotes various monitoring technologies, especially satellite observation, or other additional verification measures. Examples are the collection of and analysis of environmental samples, or the monitoring of international trade of a country. More broadly, it is understood as collecting any information that could help draw conclusions regarding a treaty partner's compliance, including potential societal verification. Although not officially acknowledged in diplomatic communities, NTM in fact means typical intelligence measures. Societal verification, and more broadly, NTM increases the probability of the detection of noncompliance and therefore constitute an important deterrent.

The Director of the International Data Centre (IDC) of the Comprehensive Nuclear Test Ban Treaty Organization, Lassina Zerbo, has also suggested that such methods be incorporated with national analysis and decision making: "When, for example, nations make a case to the commission that an on-site inspection should be launched, they would be free to supplement the information that they gather through national technical means with data collected via societal verification."

As governments continue to develop additional tools, programs and systems for passive data analysis, key questions regarding access and information flow remain:

- What legal and privacy issues need to be addressed to access data?
- How is the data integrated with other forms of NTM?
- How should states incorporate data identified and analyzed by outside experts?

NTM integration is not without its challenges. Even if the questions above are clarified and resolved, other issues remain. For instance, it is not yet clear how, and if, outside analysis is considered when states make compliance determinations. The nonproliferation community has been more successful at incorporating outside expert analysis, but input into treaty verification has not been equally established or accepted. The incorporation of active sensing also presents serious challenges if citizens from different countries participate in games, challenges or sensing that their home government may not support. NTM integration could potentially exacerbate mistrust and "big brother" fears. Given the international character of the data and tools, international norms and arrangements are more likely to foster understanding and trust, since contributing to another state's NTM may be perceived as espionage. A framework that clearly establishes the role and responsibilities of citizens, technology holders, states and international bodies is more likely to address legal concerns surrounding espionage and treason.

### 4.2 Future Collaboration and Treaty Regimes

Treaties might eventually explicitly allow for *societal* monitoring measures as part of, or in addition to NTM and other cooperative monitoring measures. Such provisions would facilitate the broader incorporation of both passive and active forms of societal verification, as well as a better defined role for outside experts. The ability to successfully integrate these tools and activities into future verification regimes will largely depend on the degree of international cooperation and collaboration early on. Such collaboration will facilitate trust and confidence in both the tools and the processes such that future agreements acknowledge an already established and proven role of society and provide for appropriate legal protections.

For such an arrangement to be effective, certain parameters must be incorporated, including:

- Non-interference provisions
- Equal access for treaty parties

This type of framework will take time to establish. Participation from non-democratic and less transparent governments will be very challenging, but is critical to the overall success of any future arrangement. Societal data and participation is inherently international in nature, and thus ultimately requires international norms to be established.

## 5. Shifting the Societal Verification Conversation

The issues and opportunities outlined above is a very general overview of the inquiry in to the value of societal verification for arms control and nonproliferation treaty verification. How to integrate information gleaned from such tools and sources into larger framework for monitoring and verifying a state's nuclear activities is a long term challenge. By clearly articulating the responsibilities of all stakeholders and establishing global norms for use and conduct, states can start to leverage the power of new information and communication technologies in a more effective and systematic manner. As a first step, the societal verification conversation needs to shift from *whether* states should take advantage of such tools and insight, to *how* they should do so.

## 6. Acknowledgements

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# MANAGING THREATS FROM EMERGING TECHNOLOGIES: CAN SAFEGUARDS SHOW THE WAY?[1]

Teri N. Leffer

National Nuclear Security Administration
U.S. Department of Energy
1000 Independence Avenue S.W., Washington DC 20585
E-mail: teri.leffer@nnsa.doe.gov

## Abstract

The system of international nuclear safeguards implemented by the International Atomic Energy Agency (IAEA) is primarily a means of verification of states' commitments under various legal instruments, principally the Nuclear Non-Proliferation Treaty (NPT), to utilize controlled nuclear fission for peaceful purposes only. However, the safeguards system can also be seen as a mechanism through which states acted to reduce the threat posed by a new technology that had a transformative impact on existing national security paradigms when it emerged in the twentieth century. In the twenty-first century, new technologies with equally profound national security implications are emerging. These include biotechnology and synthetic biology, nanotechnology, information technology, cognitive science, robotics and artificial intelligence. Throughout its history, the safeguards system has evolved to accommodate new technologies, new undertakings and new threats. Because multiple emerging technologies now constitute potential national security threats, it is appropriate to consider whether and how the lessons and successes of the safeguards system, including its capacity to evolve in response to changing requirements, could be leveraged to mitigate the threat posed by these new technologies. This paper addresses the possibility of re-imagining safeguards in a way that makes them applicable to a broader range of technology-based threats without compromising their effectiveness for their original purpose.

**Keywords:** technology; security; threat

## INTRODUCTION

The system of international nuclear safeguards implemented by the International Atomic Energy Agency (IAEA) is primarily a system of verification. Safeguards verify that a dual-use technology (controlled nuclear fission) is not being used for other-than-peaceful purposes in accordance with the terms of certain international legal instruments, most importantly the Nuclear Nonproliferation Treaty (NPT). However, safeguards can also be viewed more broadly as a mechanism for mitigating the threat associated with the existence and spread of a new technology that had a transformative impact on national security when it emerged in the mid-20th century.

[1] The concepts, analysis, opinions and policy options contained in this paper are those of the author and do not in any way represent policy of the U.S. Department of Energy or any other entity of the United States government.

In the early 21st century, a variety of new technologies are emerging that are poised to drastically disrupt existing national security paradigms in the same way that controlled nuclear fission did in the 20th century. These technologies are, for the most part, still in the early stages of development and are possessed by a relatively small number of states, but their perfection, dissemination and use are inevitable. How can the international community best mitigate the national security risks posed by these technologies while allowing states to reap the benefits of their peaceful application?

Throughout its history, the IAEA safeguards system has evolved to accommodate new technologies, new undertakings and new threats. Because multiple emerging technologies now constitute potential national security threats, it is appropriate to consider whether and how the lessons and successes of the safeguards system, including its capacity to evolve in response to changing requirements, could be leveraged to mitigate these new threats. This paper addresses the possibility of re-imagining safeguards in a way that makes them applicable to a broader range of technology-based threats without compromising their effectiveness for their original purpose.

**SAFEGUARDS: UNIQUE OR REPLICABLE?** The nuclear nonproliferation regime[2] restricts the behavior of participating states with respect to their use of the technology of controlled nuclear fission. States consent to such restriction in exchange for access to nuclear technology and the benefits of its peaceful application. Safeguards are the mechanism by which states' compliance with their commitment to use nuclear technology only for peaceful purposes is verified. The success of any multilateral control regime for a dual-use technology can be assessed by multiple factors, including: the degree of state participation in the regime; the regime's duration; the existence and effectiveness of a verification system for the regime; the frequency with which the regime has been successfully circumvented; the extent to which the controlled technology has proliferated despite regime restrictions; and the frequency with which the controlled technology has been used for other-than-peaceful purposes since the regime's entry into force. By these measures the IAEA safeguards system, more than fifty years after its inception, still constitutes one of the most successful example to date of a multilateral system for the management of national security risk from a dual-use technology. Safeguards are applied in virtually all states that use nuclear material and have been accepted to such an extent that they can now legitimately be considered to constitute an international norm.

The success enjoyed by the IAEA and by the international safeguards system is attributable for the most part to the political environment that pertained during the decades after controlled nuclear fission emerged as a mature technology. This environment generated the political will that was necessary for the creation of the nuclear nonproliferation regime, including safeguards, and for its effective implementation up to the present day. But is political will alone sufficient to support an effective verification regime for a dual-use technology? The existence of well-subscribed international agreements for the control of other dual-use technologies that lack a corresponding verification mechanism (the Biological and Toxin Weapons Convention being the

[2] For purposes of this analysis, the nuclear nonproliferation regime is considered to consist of the Nuclear Nonproliferation Treaty (NPT), mandatory safeguards agreements entered into by non-nuclear weapons states parties to the NPT, voluntary safeguards agreements entered into by nuclear weapons states, associated export control regimes (e.g., the Zangger Committee and the Nuclear Suppliers Group) and associated resolutions of the United Nations Security Council (e.g., UNSCR 1640).

foremost example) suggests that there are separate thresholds for each component: in other words, the political will that is sufficient to generate a broadly-subscribed multilateral agreement for control of a given technology may not be enough to establish an effective verification mechanism for that agreement. This is due, in part, to the technology's inherent characteristics, which play a major role in either facilitating or thwarting the creation of an effective verification system for any such agreement. They can be a headwind against which the verification system must continually push, or a tailwind that helps propel it to success. If this is the case, then the IAEA safeguards system owes at least some of its success to aspects of the technology of controlled nuclear fission itself, aspects which may or may not pertain to other technologies for which international control regimes have been or will be established. If so, then it would be useful to identify those characteristics of controlled nuclear fission which made it amenable to an effective verification system and to determine whether any of the emerging technologies of the early 21st century exhibit those same characteristics. To the extent that they do, these technologies may be good candidates for a treaty-based international control regime incorporating a safeguards-like system of verification.

Evaluating the potential of emerging technologies to support a system of verification and controls similar to IAEA safeguards requires a brief digression into the origin, development and impact of new technologies. We may return to the discussion of whether and how the principles and techniques of safeguards may be applied to these technologies.

## CHARACTERIZING TECHNOLOGIES FOR INHERENT SAFEGUARDABILITY

Technologies can be characterized in any number of ways: according to their stage of development, their degree of dissemination and adoption, their economic and societal impact, and so forth. For purposes of assessing the amenability of technologies to a verification system, it is useful to characterize them according to three attributes:

**Applicability.** Does a technology have only a few realized or potential applications, or a very large number of applications? Technologies that are more generally applicable spread rapidly throughout the economy and society and are therefore harder to control, all other factors being equal. The applicability of technology can be described along a continuum from those which have only a few uses, such as hydraulic fracturing technology used by the oil and gas industry, to those which have near-universal applicability, such as writing and agriculture (see Figure 1).

Technologies which have the broadest applicability are called *general-purpose technologies* or GPTs. A GPT is a new method of producing and inventing that is significant enough to have a deep and protracted impact on the economy and on society as a whole. A GPT is pervasive, improves over time (thereby lowering its cost to users) and makes it easier to invent or produce new products or processes.[3] Experts differ in their assessment of which technologies constitute GPTs, but it has been suggested that over the course of human history there have been only

[3] See Philippe Aghion and Steven N. Durlauf, eds., *Handbook of Economic Growth, Volume 1B.* Elsevier, 2005.

twenty-four technologies that can be classified as true GPTs, among them the domestication of plants and animals, the wheel, writing and the internet.[4]

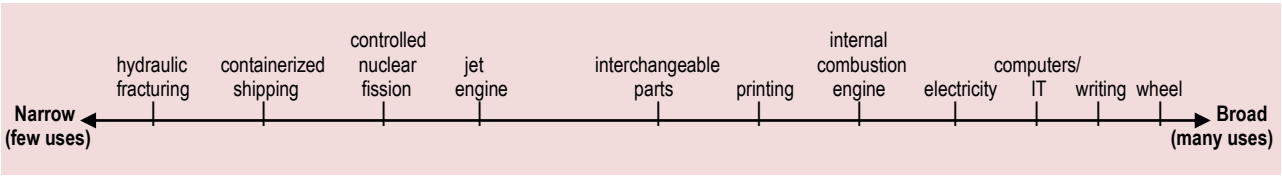



***Figure 1.*** *Spectrum of technology applicability*

**Cost.** How expensive is it to develop, acquire and use a particular technology? The cost of a given technology plays a part in that technology's attractiveness versus alternatives and helps to determine the speed and extent of its spread. Technologies which are inexpensive to use, even if the initial development and acquisition costs are high, are more attractive and should, on balance, be harder to safeguard than technologies that are costly to acquire and that remain costly to use relative to their alternatives. The very high initial cost of controlled nuclear fission (i.e., the cost to acquire nuclear material) contributes substantially to its safeguardability. The fact that acquiring nuclear material is the most costly and difficult element of the technology is one of the reasons why nuclear material serves as the leverage point for the majority of safeguards techniques, which are designed to account for material and detect its diversion.

**Complexity.** Are applications of the technology usable by non-specialists, or is some sort of training required in order to use them? If the latter, how difficult is it to master? Technologies that are accessible to anyone should be harder to control than those which are highly complex and require a specialized skill set in order to use. The use of controlled nuclear fission, regardless of whether it is being applied for power generation, propulsion, weapons, source production, or any other application, requires specialized training and is therefore inherently more safeguardable than a technology which is usable by anyone without the need for any sort of prior training.

With these three criteria, it is possible to construct a representation of the "technology space" for safeguards which can help to categorize technologies and identify those which are the most or least amenable to a safeguards-like system of verification and control (see Figure 2). All other factors being equal, the easiest technologies to safeguard should be those technologies that have only a few applications and are therefore not widely dispersed throughout society and the economy, have a high cost barrier to their acquisition and use, and have a high complexity barrier to their use. Controlled nuclear fission satisfies all of these conditions and is therefore a technology with a high inherent safeguardability. The most difficult technologies to safeguard, again assuming all other factors are equal, should be those technologies that have many actual or potential applications, are inexpensive to acquire and use, and which require no specialized

[4] The technology of controlled nuclear fission does not meet the generally accepted criteria for a GPT, which are: 1) it presents as a single, recognizable generic technology; 2) it has much scope for improvement initially but comes to be widely used across the economy; 3) it has many different uses; and 4) it creates many spillover effects. See Lipsey, Richard; Kenneth I. Carlaw and Clifford T. Bekhar, *Economic Transformations: General Purpose Technologies and Long Term Economic Growth*. Oxford University Press, 2005, pp. 131–218.

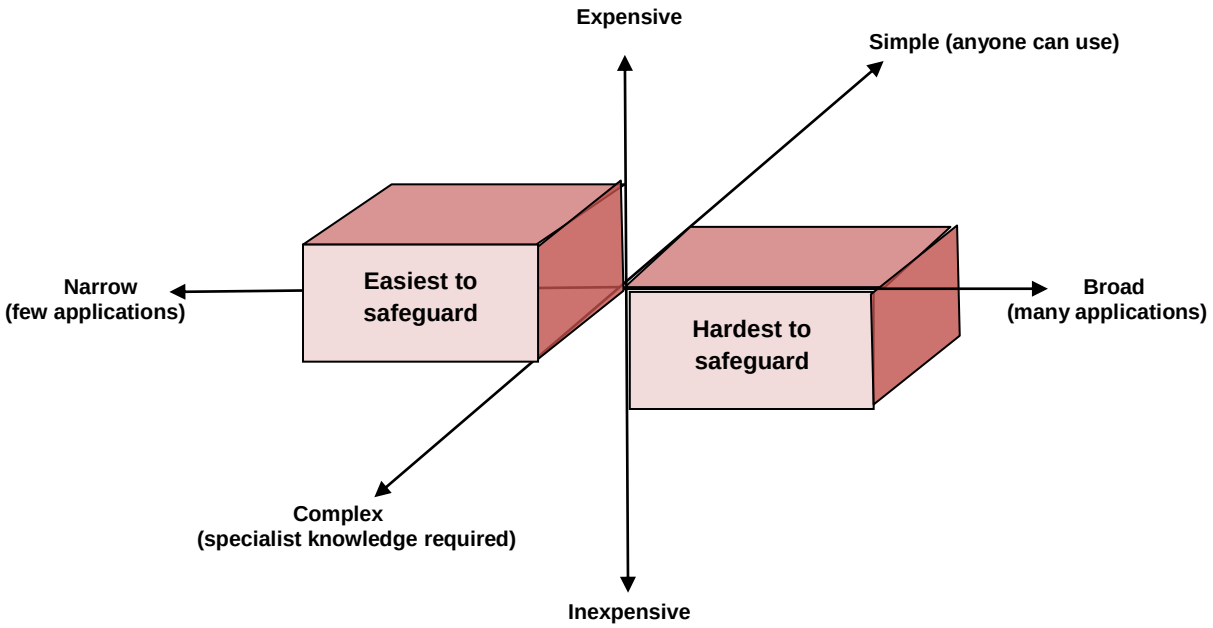



***Figure 2.*** *Inherent safeguardability of technologies*

training for their use. General-purpose technologies typically satisfy all three of these criteria, making them inherently difficult to safeguard.

**EMERGING TECHNOLOGIES.** Emerging technologies constitute significant technological advances that render accessible far-reaching innovations in their respective fields. Emerging technologies can be categorized according to their degree of development. Many organizations use the concept of *technology readiness level* (TRL) to characterize emerging technologies. This paper characterizes technologies according to the U.S. Department of Energy TRL system, which recognizes nine levels of technological maturity (see Figure 3). For purposes of this study, which attempts to place emerging technologies into a political and national security context rather than a purely scientific one, a technology is considered to be emerging -- that is, becoming feasible to the point where governments must begin paying attention to potential consequences of its applications -- when it reaches TRL 7 (a prototype of the technology is at or near the level of the planned operational system). A technology is considered to have fully emerged when it reaches TRL 9 (actual application of the technology in its final form).

## THE TECHNOLOGY OF CONTROLLED NUCLEAR FISSION

Nuclear fission and the related potential for the chain reaction were discovered in 1938, leading to concern that the Nazi government in Germany would become aware of this new physical principle and its potential to be exploited for weapons purposes. Prompted by warnings from prominent physicists, the United States began work in 1939 on what would, in 1942, become the Manhattan Project, with the goal of realizing the weapons potential of controlled nuclear fission before Germany could do so. As part of the Manhattan Project, the first controlled nuclear chain reaction was achieved at the University of Chicago in December 1942. At this point, the

technology of controlled nuclear fission[5] was at approximately TRL 3 (initiation of active research and development). With the Trinity test in July of 1945, it had reached TRL 8 (technology qualified through test and demonstration). One month later, the atomic bombings of Japan marked the final emergence of controlled nuclear fission as a mature technology and demonstrated on the largest stage imaginable the catastrophic nature of the threat posed by its weaponization.

| Technology Readiness Level | Description |
|---|---|
| **TRL 1** | **Scientific research moves to applied research and development (R&D):** Lowest level of technology readiness. Examples include paper studies of a technology's basic properties. |
| **TRL 2** | **Invention begins:** Once basic principles are observed, practical applications can be invented. Applications are speculative and there may be no proof or detailed analysis to support the assumptions. Examples are limited to analytic studies. |
| **TRL 3** | **Active R&D is initiated:** This includes analytical studies and laboratory studies to physically validate analytical predictions of separate elements of the technology. Examples include components that are not yet integrated or representative. |
| **TRL 4** | **Basic technological components are integrated:** Technological components are integrated to establish that the pieces will work together. |
| **TRL 5** | **Fidelity of breadboard technology improves significantly:** The basic technological components are integrated with reasonably realistic supporting elements so it can be tested in a simulated environment. Examples include high fidelity laboratory integration of components. |
| **TRL 6** | **Model/prototype is tested in a relevant environment:** Representative model or prototype system, which is well beyond that of TRL 5, is tested in a relevant environment. Represents a major step up in a technology's demonstrated readiness. Examples include testing a prototype in a high-fidelity laboratory environment or in a simulated operational environment |
| **TRL 7** | **Prototype near or at planned operational system:** Represents a major step up from TRL 6, requiring demonstration of an actual system prototype in an operational environment. |
| **TRL 8** | **Technology is proven to work:** Actual technology completed and qualified through test and demonstration. |
| **TRL 9** | **Actual application of technology in its final form:** Technology proven through successful operations. |

***Figure 3.*** *Technology Readiness Levels in the U.S. Department of Energy*

Three months after the bombings of Japan, the states that had collaborated under the Manhattan Project (the United States, the United Kingdom[6] and Canada) issued the Three State Declaration

[5] For purposes of this study, the development of the technology of controlled nuclear fission refers to its first application (weaponization) rather than to power generation, propulsion or other subsequent applications.

[6] The United Kingdom was initially reluctant to collaborate with the United States on atomic research since its own atomic project, code named Tube Alloys, was more advanced than the U.S. program and the British were loath to share their technological lead. Only after it became clear that the U.S. research effort was poised to surpass that of the U.K. was an offer to collaborate forthcoming. The U.S., having taken the lead in atomic research, subsequently restricted the flow of information to the British in order to prevent the U.K. from being able to build its own atomic weapons after the war. This episode is instructive as to how states are likely to approach the prospect of sharing control over any newly emerging technology.

on Atomic Energy, which required effective safeguards and inspections as a precondition for access to peaceful applications of controlled nuclear fission. At this point, all three governments were considering placing atomic weapons under international control. In 1946 the United States established a Committee on Atomic Energy that was charged with identifying a workable international arrangement that would promote the peaceful uses of controlled nuclear fission while preventing its weaponization. The committee's final report, which became known as the Acheson-Lilienthal Report, acknowledged the inevitable spread of the new technology and recommended creating a system of international control to govern its use.

Later that year, the newly-created United Nations Atomic Energy Commission began to discuss how an international control system for nuclear technology might be set up. The plan put forward by the United States (known as the Baruch Plan after Bernard Baruch, the U.S. representative to the Commission), called for centralized international control over nuclear technology and nuclear material production. But the Soviet Union, which by then was working on its own bomb, was unwilling to accept the establishment of an international control authority for atomic weapons and technology before the United States had relinquished its own atomic weapons. Thus, the notion of centralized international control over nuclear technology arrived stillborn into an as-yet unproliferated world in which a single state held a monopoly on the new technology that other states were determined to break. This took place in 1949 with the first Soviet nuclear test. The United Kingdom followed with its own test in 1952, and in 1953 the first commercial application of nuclear energy was realized as construction began on the Calder Hall nuclear power reactor in the United Kingdom. Collectively, these developments gave fresh impetus to the search for a workable means of international management of nuclear technology, and the result was President Eisenhower's "Atoms for Peace" proposal issued later that year. That proposal initiated the negotiation of the IAEA statute, which led to the establishment of the IAEA and its initial system of safeguards in 1957.

**CHARACTERISTICS OF CONTROLLED NUCLEAR FISSION.** In retrospect, several aspects of the technology of controlled nuclear fission and the context in which it developed to maturity stand out as being significant for the eventual establishment and continuing success of the multilateral system of controls for that technology as constituted by the IAEA and the system of international safeguards.

**Inherent characteristics.** The technology of controlled nuclear fission has a number of inherent characteristics that make it amenable to a system of safeguards designed to restrict its use to peaceful purposes. It has a narrow range of potential applications; it has an obvious weapons application; its use requires specialist knowledge and training; it is dependent on a critical material that is difficult and expensive to acquire and hard to acquire clandestinely; and it exhibits a number of natural leverage points for controls.

> ***Applicability.*** The most widely-applicable technologies are the general purpose technologies, which have so many applications that they come to permeate the economy and society as a whole. In many ways, nuclear technology is the opposite of a general-purpose technology, which makes it ideally suited to safeguards. Since its emergence in 1945, controlled nuclear fission has had only a handful of applications: the generation of electric power, the creation of isotopes for medical purposes, naval propulsion, and the fabrication of extremely powerful

explosive devices.[7] There have been attempts to find other applications for controlled nuclear fission -- for example, the U.S. and Soviet atomic demolition munitions programs and the U.S. Air Force's Nuclear Energy for the Propulsion of Aircraft (NEPA) program -- but in each case nuclear technology proved to be less suitable than the available alternatives. The relatively narrow range of applications for nuclear technology is a key factor in its amenability to multilateral controls on its use.

When considering applicability, there is one application which outweighs all others: the potential for the new technology to be used as a weapon. The *raison d'être* for international management of controlled nuclear fission is to prevent states from using the technology to acquire weapons. An emerging technology that lacks an obvious weapons application is unlikely to find itself the subject of a drive to establish international control over its use. Another critical element is the actual or imminent realization of commercial uses for the new technology. The fact that the Calder Hall atomic power plant was already under construction by 1953 was a significant factor leading to the adoption of the IAEA statute in 1957. Thus, questions we should ask with respect to an emerging technology's applicability include:

- Is the new technology a general-purpose technology?
- How broad is the range of potential applications for the new technology?
- Do the actual or potential applications of the new technology include weapons that are novel or that are of significantly greater military utility than existing weapons?
- Are there existing or imminent commercial applications for the new technology at the time of its emergence?

***Complexity.*** The complexity of a new technology's applications[8] plays an important part in determining the speed with which the technology spreads throughout society as well as the new technology's attractiveness relative to alternatives. Complexity manifests itself practically as a skill barrier that must be overcome in order to use the various applications of a technology. For example, the skill barrier to using the applied technology of language (learning to read and write) is surmountable by most people in early childhood. Likewise, the skill barrier to using the applied technology of the internal combustion engine (learning to drive) is also readily surmountable by most people. In contrast, the skill barrier to using the applied technology of heavier-than-air aircraft (learning to fly an airplane) is surmountable by fewer people, while the skill barrier to using controlled nuclear fission in any application (power generation, propulsion, weapons) is surmountable by still fewer people. For any emerging technology, then, a key factor affecting its safeguardability is the complexity of its application:

[7] Although there are other uses for nuclear technology apart from these principal ones (for example, nuclear technology is used in blood irradiators, has been used as part of insect eradication programs, etc.), these uses are too few and too specialized to constitute a serious obstacle to the establishment of an international system for control over nuclear technology.

[8] It is the complexity of a new technology's applications, and not that of the technology itself, which constitutes the barrier to its use and spread. It isn't necessary to know how to construct an internal combustion engine in order to operate a motor vehicle, nor does one need a thorough understanding of hydraulics to be able to flush a toilet.

- Are applications of the new technology accessible to anyone who wishes to use them with little or no training, or is specialized training required? If so, how high a barrier does that specialized training constitute?

***Cost.*** The context in which controlled nuclear fission was developed into a mature technology was the Manhattan Project. The research effort that would become the Manhattan Project began modestly in 1939 but grew into a massive state-sponsored effort that employed more than 130,000 people and cost nearly $2 billion (equivalent to about $26 billion today).[9] The effort required the construction of two enormous physical plants (the uranium enrichment facility at Oak Ridge and the plutonium production facility at Hanford) and a secret city in New Mexico to turn the products of the Oak Ridge and Hanford facilities into weapons. Over 90% of the cost of the Manhattan Project was for construction of factories and the production of fissionable materials, with less than 10% for the development and production of the weapons themselves.[10] The return on this unprecedented effort was the evolution of controlled nuclear fission from TRL 3 to TRL 9 in a span of only three years. The expense and logistical difficulty of producing fissionable materials clearly constitutes a formidable barrier to acquiring nuclear technology. For newly-emerging technologies, key questions related to cost are:

- Is the technology or its critical component expensive to acquire relative to its alternatives?
- Do the costs of the technology diminish significantly once the initial infrastructure or material investment has been made?

***Leverage points for controls.*** The technology of controlled nuclear fission depends on a critical material that is difficult and expensive to obtain and whose creation requires the construction of a substantial physical infrastructure that is hard to conceal. In addition, the key material possesses a physical property (the emission of radiation) that manifests in a consistent manner, is well understood, and which can be detected remotely and measured. Each of these factors constitutes a separate leverage point that can be (and is) exploited by a system of safeguards designed to ensure that the technology is used only in acceptable ways. For emerging technologies, we may therefore ask:

- Is the technology dependent on a critical material such that the inability to create or obtain this material makes it impossible to utilize the technology?
- Does it require the construction of a large, hard-to-conceal physical infrastructure?
- Does it exhibit any physical property that can be leveraged for a system of controls (i.e., measured, detected remotely, etc.)?

[9] Nichols, Kenneth D. *The Road to Trinity: A Personal Account of How America's Nuclear Policies Were Made.* New York: William Morrow and Company, 1987, pp. 34-35

[10] "Atomic Bomb Seen as Cheap at Price," (http://news.google.com/newspapers?id=yuVkAAAAIBAJ&sjid=KoENAAAAIBAJ&pg=5621%2C2841878) Edmonton Journal, August 7, 1945, retrieved March 31, 2013

- Does it incorporate any material(s) that can be reliably sampled and analyzed for evidence of prohibited activities?

**Political context.** Regardless of how much a technology's inherent characteristics help or hinder its safeguardability, the political environment into which the new technology emerges will determine whether, how, and how quickly an international system of controls on its use will be established. Key factors include the degree to which control of the technology is concentrated during its emergence and the existence or absence of a "demonstration effect" event involving that technology.

***Concentration of control.*** Controlled nuclear fission was developed into a mature technology by the government of a single state, as opposed to simultaneous independent efforts undertaken by multiple states or by private entities. The work was undertaken in absolute secrecy, without the potential for either collaboration as in an academic environment or technology theft as in private industry,[11] in a wartime effort of unprecedented scope and expense as opposed to a resource-constrained academic or industry environment. At the time of its emergence, the mature technology was entirely under the control of the government of a single state. Once it had emerged, it was clear that the United States would not keep its monopoly on nuclear technology forever, that commercial applications of the new technology were imminent, and that the establishment of a system of international controls on its use was therefore in the United States' interest. Thus, important questions to ask with respect to an emerging technology include:

- Is the technology being developed exclusively by government(s) as opposed to an academic or private industry environment?
- Is the technology being developed by a single entity or by multiple entities engaged in simultaneous independent efforts?
- How many entities (both states and non-state actors) possess the technology at the time of its emergence?
- How many entities appear likely to possess it within the first few years after its emergence?

***Demonstration effect.*** The political impetus for the development of an international system of controls on the use of nuclear technology benefited enormously from the demonstration effect constituted by the atomic bombings in Japan. After August 6, 1945, the existence, technical viability and destructive impact of nuclear weapons were no longer matters of conjecture or open to debate. Thus, an important factor in determining whether emerging technologies could be subject to multilateral control is whether or not the national security implications of the new technology have been publicly demonstrated:

- Has a weapon that is clearly based on the new technology been used overtly?

[11] Although Soviet spies did infiltrate the Manhattan Project, the impact of their activities on the speed with which nuclear technology spread was minimal since the principal constraint on the Soviet atomic program was a shortage of uranium ore. See Holloway, David, *Stalin and the Bomb: The Soviet Union and Atomic Energy, 1939-1956.* New Haven, CT, Yale University Press, 1994, pp. 222-223.

- Has the technology been used by terrorists?
- Has a widely-publicized accident involving the technology taken place?

The characteristics exhibited by the technology of controlled nuclear fission from the time that its development began in earnest (the startup of what would become the Manhattan Project in 1939) to the creation of a regime of controls on its use by governments (the entry into force of the IAEA statute in 1957) are summarized in Figure 4 below.

| Emerging Technology Attributes | | | Controlled Nuclear Fission (1939-1957) |
|---|---|---|---|
| **Inherent characteristics** | Applicability | General purpose technology (GPT)? | No |
| | | Broad range of potential applications? | No (weapons, power generation, propulsion) |
| | | Existing or imminent weapons applications? | Yes (atomic bomb) |
| | | Existing or imminent commercial application? | Yes (power generation) |
| | Complexity | Specialized knowledge/training required to use? | Yes |
| | Cost | Critical material expensive to create/obtain? | Yes (fissile material) |
| | | Cost of technology diminishes significantly once initial investment has been made? | Yes (most expensive aspect of nuclear technology is obtaining fissile material) |
| | Leverage points for controls | Exhibits property that can be detected remotely? | Yes (radiation) |
| | | Dependent on single critical component or materials? | Yes (fissile material) |
| | | Components or materials that can be sampled and analyzed for evidence of prohibited activity? | Yes (HEU, Pu, others) |
| | | Requires construction of large, hard to conceal physical plant? | Yes |
| **Political context** | Concentration of control | Technology developed exclusively by government(s)? | Yes |
| | | Technology developed by a single entity or by multiple entities engaged in simultaneous, independent efforts? | Single entity (United States government) |
| | | Number of entities possessing technology at time of emergence | One (United States government) |
| | | Number of entities that appear likely to possess technology within first few years after its emergence | Several (Soviet Union, UK, Canada) |
| | Demonstration Effect | Overt weaponization? | Yes (atomic bombings of Japan) |
| | | Terrorist use? | No |
| | | Accident? | No |

***Figure 4.*** *Emerging technology characteristics of controlled nuclear fission, 1939-1957*

## EMERGING TECHNOLOGIES WITH NATIONAL SECURITY IMPLICATIONS

Nearly every technology has national security implications ranging from the profound to the negligible, and as the twenty-first century unfolds and the rate of technology development accelerates, it is certain that the international community will be confronted with a variety of new, technologically-driven security challenges. But it is also certain that some of these new capabilities will have more immediate and more profound national security impacts than others: for example, the perfection of a technology to grow artificial meat *in vitro* would have fewer and less direct potential national security implications than would the perfection of either swarm robotics or laser propulsion. The challenge for policymakers is to identify which technologies

have the greatest potential to disrupt existing national security paradigms and then to devise and implement appropriate mitigation measures in a timely fashion.

Over the last decade, the inherent potential of new technologies to cause national security disruptions has received a great deal of attention. Multiple studies, reports and conferences have examined the national security implications of emerging technologies[12]. Taken collectively, this body of thought reveals a growing consensus as to which of the technologies that are now emerging are likely to have the most immediate and profound impact on existing national security paradigms. The technologies that are cited most frequently include nanotechnology, biotechnology, robotics, information technology, cognitive science and artificial intelligence. A consideration of all of these technologies is beyond the scope of this paper. Instead, this analysis will concentrate on four technologies that collectively represent the range of early 21st century innovation: molecular manufacturing, synthetic biology, robotics and information technology. These are the technologies which will be examined for their potential to benefit from the lessons and techniques of IAEA safeguards. The selected technologies are described briefly below, along with an assessment of their current readiness levels, their degree of dissemination, their potential national security impacts and any existing proposals for their control.

**MOLECULAR MANUFACTURING.** Molecular manufacturing, sometimes referred to as molecular nanotechnology, is the most advanced form of nanotechnology, an emerging technology that involves the discovery and exploitation of novel behaviors and properties of materials with dimensions between 1 and 100 nanometers[13] (referred to as nanomaterials). Like the steam engine, electricity and computers, nanotechnology is a general purpose technology with the potential to drastically alter society through its impact on existing economic and social structures. When fully developed, nanotechnology holds the potential for vastly improved manufacturing processes which could make the mass production of physical objects as inexpensive as copying a computer file. The most advanced nanotechnology, so-called fourth generation nanotechnology or molecular manufacturing, will enable the chemical synthesis of complex structures by mechanically positioning reactive molecules, allowing for the assembly of nanoscale machines. Among the items that such "nanofactories" could produce would be more nanofactories, thereby multiplying exponentially the productive capacity of the technology. Once mature, the technology of molecular manufacturing will have a societal impact as profound as that of the industrial revolution but compressed into a timescale of a few years rather than six decades. Molecular manufacturing is presently in the very earliest stages of development (TRL 1), and its emergence as a mature technology is generally assessed to be several decades away.

The national security implications of molecular manufacturing include:

[12] See, for example, Carolyn S. Mattick, Brad R. Allenby and George R. Lucas, Jr. *Implications of Emerging Military/Security Technologies for the Laws of War*, 2012 Chautauqua Council Final Report. Additional examinations of the national security implications of emerging technologies may be found in reports of the Consortium for Emerging Technologies, Military Operations and National Security (CETMONS, http://lincolncenter-dev.asu.edu/CETMONS/) and the Center for Responsible Nanotechnology (http://www.crnano.org).

[13] By comparison, a human hair is roughly 100,000 nanometers wide.

- cheap, easily produced nanoscale weapons and surveillance devices[14]
- small, portable nanofactories that can be easily smuggled
- desktop manufacturing of weapons and surveillance devices[15]
- use of nanomanufacturing by terrorists, both individuals and groups
- malicious release of free-range self-replicators[16]

Nanotechnology is still in its infancy and is not yet broadly disseminated. Although many states have launched nanotechnology research initiatives,[17] no government has yet offered a proposal for international controls on molecular manufacturing or on nanotechnology research or applications in general.

**SYNTHETIC BIOLOGY.** Synthetic biology is a subset of biotechnology that incorporates elements of engineering to enable the design and construction of devices that use biological systems as their platform. Synthetic biology differs from genetic engineering in that it does not merely alter the DNA of existing organisms, but instead uses identified gene sequences as building blocks for the construction of entirely new organisms that exhibit some desired useful combination of traits not found in nature. The process is similar to computer programming in that it involves the assembly of blocks of genetic code into sets of instructions for cellular machines. Potential applications of synthetic biology range from human health (new vaccines, pharmaceuticals and diagnostic tools) to energy (carbon-neutral alternative fuels) and the environment (organisms engineered to consume pollutants).

[14]The latter are particularly problematic, since governments will make decisions under the assumption that adversaries are monitoring much of what they are doing, thereby contributing to crisis instability. "Technology and Innovation 2025," Toffler Associates, November 2008 , p. 7 (http://www.toffler.com/docs/2025.pdf, accessed February 27, 2013)

[15] Desktop manufacturing is already commercially available, although not via molecular manufacturing but rather by additive manufacturing, in which successive layers of material are laid down in varying shapes according to a digital template. Desktop devices for additive manufacturing (3-D printers) can be purchased for less than USD 1,000 per unit and are capable of turning any .STL file downloaded to its software into a three-dimensional plastic object (including, to great Internet acclaim, a working handgun: http://www.forbes.com/sites/andygreenberg/2013/05/03/this-is-the-worlds-first-entirely-3d-printed-gun-photos/)

[16] When nanotechnology-based manufacturing was first conceived, a concern arose that tiny manufacturing systems might be accidentally introduced into the environment and 'eat' the biosphere, reducing it to copies of themselves (a scenario referred to as ecophagy or "grey goo"). Although it is now clear that replicating assemblers will not be used for manufacturing, grey goo remains a risk. Grey goo has essentially no military or commercial value and only limited terrorist value, but could be used as a tool for blackmail: cleaning up a single grey goo outbreak would be very expensive and might require severe physical disruption of the affected area (the nanotechnology equivalent of a radioactive "dirty bomb"). Another possible source of grey goo release is irresponsible or attention-seeking hobbyists (the nanotechnology equivalent of computer worms and viruses). Center for Responsible Nanotechnology, http://www.crnano.org/dangers.htm, accessed March 17, 2013.

[17] The United States established its National Nanotechnology Initiative (http://www.nano.gov) in 2000. Between 2001 and 2004, more than 60 countries established their own national-level nanotechnology programs (Sargent, John F. "Nanotechnology: A Policy Primer," Congressional Research Service, April 13, 2012, p. 4). However, only a few countries are acknowledged as leaders in nanotech research and development, among them the United States, Japan, Germany, South Korea and Taiwan (Hwang, David, "Ranking the Nations on Nanotechnology," *Solid State Technology*, August 27, 2010, http://www.electroiq.com/articles/stm/2010/08/ranking-the-nations.html, accessed April 30, 2013).

The accelerating development of synthetic biology is made possible by the ever-increasing speed and diminishing cost of key enabling technologies, such as DNA sequencing, gene fabrication and computer modeling of how synthetic genes will behave. Synthetic biology has progressed to the extent that it is now cheaper to synthesize a gene that it is to clone it.[18] In addition, the large and growing collection of genome databases provides a readily-accessible source of templates for the creation of new viruses at minimal cost.

Potential national security implications of synthetic biology include:

- synthesis of treatment-resistant bacteria and viruses to serve as bioweapons
- synthetic organisms designed to generate weapons-usable materials or components (e.g., explosives, propellants, etc.)
- use of synthetic organisms by terrorists
- accidental release into the environment of synthetic organisms that may prove harmful or difficult to eradicate ("bioerror" as opposed to bioterror)

Synthetic biology is a newly-mature but still evolving technology, having reached TRL 9 in 2010 with the creation of the first entirely artificial living organism.[19] Several U.S. government agencies are funding research in synthetic biology and the European Union is funding the development of a European strategy for synthetic biology, but the majority of the considerable funding being directed at this area comes from the commercial private sector. Although leadership in the research and development of biotechnology and synthetic biology is confined to a few states (with the U.S. spending by far the most money per year on R&D[20]), many states are actively pursuing synthetic biology research, including Canada, India, Israel, Japan, Korea, Slovenia and Turkey.[21] Although it is very new, the technology of synthetic biology is nevertheless widely disseminated due to the commercial nature of much synthetic biology research. The standard, interoperable pieces of DNA that are used for creating synthetic cellular machines and organisms, known as "biobricks," are already widely commercially available,

[18] In 2007, several companies offered gene sequence synthesis up to 2,000 base pairs long for a price of about $1 per base pair and a turnaround time of less than two weeks (Pollack, Andrew, "How Do You Like Your Genes? Biofabs Take Orders," http://www.nytimes.com./2007/09/12/technology/techspecial/12gene.html?pagewanted=2&_r=1 ). As of May 2013, the price had dropped to less than $0.29 per base pair with a turnaround time as low as four days (http://www.genscript.com/gene_synthesis.html?src=google&gclid=CLLA7uis-rYCFcIv4Aod-B0ACQ, accessed May 3, 2013).

[19] The J. Craig Venter Institute created the organism by synthesizing the entire genome of one bacterium and transplanting it into another, an undertaking that required fifteen years and $40 million. At a news conference, Venter described the new organism as "the first self-replicating species…on the planet whose parent is a computer." "First Synthetic Organism Created," http://discovermagazine.com/2011/jan-feb/02 , accessed May 3, 2013.

[20] Van Beuzekom, Brigette and Anthony Arundel. *OECD Biotechnology Statistics 2009*, p. 25 (http://www.biotechnologie.de/BIO/Redaktion/PDF/de/laenderfokus/suedkorea-oecd-report,property=pdf,bereich=bio,sprache=de,rwb=true.pdf, accessed May 3, 2013)

[21] Caruso, Denise. "Synthetic Biology: An Overview and Recommendations for Anticipating and Addressing Emerging Risks, *Science Progress*, 2008, p. 4 http://www.bio.org/articles/synthetic-biology-overview-and-recommendations-anticipating-and-addressing-emerging-risks, accessed May 3, 2013.

accessible to anyone with an Internet connection and a credit card, and the relevant skills for creating them are known to any reasonably competent biology graduate student.[22]

Applications of synthetic biology that are expressly conceived as weapons are already prohibited under the Biological and Toxin Weapons Convention (BWTC), but the status of other weapons-related applications is murkier.[23] During the most recent BWTC Review Conference in 2012, delegates acknowledged that the speed of advances in biotechnology made review of new developments necessary on an annual basis as opposed to every five years as is now the case. As of yet, no government has offered proposals for new controls pertaining specifically to synthetic biology.

**ROBOTICS.** Much attention has been paid in recent years to the emergence of unmanned and robotic weapons, in particular unmanned aerial vehicles (UAVs) or drones, as a result of their often controversial use by the United States in Iraq, Afghanistan and elsewhere. In 2001, the United States Congress set a goal for the Army, stating "…that, within 10 years, one-third of U.S. military operational deep strike aircraft would be unmanned, and, within 15 years, one-third of all U.S. military ground combat vehicles would be unmanned."[24] Unmanned and robotic systems are attractive because of their military utility: they allow combat to be conducted over a larger geographical area and reduce human casualties by keeping soldiers out of harm's way. They are capable of integrating and processing much larger quantities of information much more rapidly than a human soldier could do. They are not subject to fatigue or psychological stress. Robotic weapons also act as a force multiplier, since their deployment reduces the number of troops required for a given mission. This aspect could incentivize continual increases in robotic systems' level of autonomy, since ongoing, labor-intensive human oversight would dilute the force multiplication effect. On the other hand, a robotic system has no need to protect itself and may therefore act more conservatively than a human would in making decisions regarding use of force.[25]

Most of the unmanned and robotic weapon systems that have been deployed to date are simply an extension of the soldier, meaning that a human remains in control at all times. They are capable of moving and targeting autonomously, but a human must still make the decision to fire. These systems include some types of torpedoes, Predator and Reaper UAVs, and Israel's Harpy missile.[26] However, an increasing number of robotic systems are capable of being programmed

[22] Moreno, Jonathan D. "Synthetic Biology Grows Up," Science Progress, http://scienceprogress.org/2010/05/synthetic-biology-grows-up, accessed May 3, 2013.

[23] Applications of synthetic biology that are weapons-related but not weapons themselves, such as synthetic organisms that produce explosives, could conceivably be seen as violating Article 1 of the BTWC, which enjoins signatories to neither produce nor possess microbial or other biological agents "that have no justification for prophylactic, protective or other peaceful purposes." Carlson, Rob and Daniel Grushkin, "The Military's Push to Green Our Explosives," Slate, January 19, 2012, http://www.slate.com/articles/technology/future_tense/2012/01/synthetic_biology_environmentally_friendly_weapons_and_the_biological_and_toxin_weapons_convention_.html, accessed May 5, 2013.

[24] "U.S. Army Roadmap for Unmanned Systems: 2010-2035," http://www.fas.org/irp/program/collect/uas-army.pdf (September 21, 2012), p. 5, accessed May 1, 2013.

[25] Mattick, et. al., p. 16-18

[26] ibid., p. 17

to act with varying degrees of autonomy. Such systems could, in principle, make their own decisions regarding the application of force, including lethal force. An example of such an autonomous system is iRobot's Packbot. Most models of the Packbot are capable of Tasering enemy combatants without the need for authorization from a human decision-maker, but some models are equipped with lethal weapons.[27] Other examples of potentially lethal autonomous systems include the Samsung SGR-A1 robot, which is used by South Korea to patrol the demilitarized zone at the border with North Korea, as well as the U.S. Navy's Patriot and Phalanx missiles. In practice, these systems are all presently supervised at some level by human operators and their software includes limits on which decisions can be delegated to the computer. But in principle, each of these systems could be capable of making an autonomous decision to use lethal force.

The national security implications of robotics include:

- autonomous weapons capable of making decisions to apply lethal force without a "human-in-the loop"[28]
- proliferation of lethal autonomous systems to terrorist groups
- vulnerability of lethal autonomous systems to cyber attack or hijacking

As of today, there is no technological barrier to the construction and deployment of the types of systems described above, including lethal autonomous variants. More than seventy countries now possess unmanned aerial vehicles (UAVs) and pilotless aircraft, but only five (the United States, the United Kingdom, China, Israel and Italy) operate UAVs that are armed.[29] Other types of robotic weapons systems are currently deployed by the United States, Israel, South Korea, Japan and Singapore. More sophisticated unmanned and robotic systems, such a swarm robotics, are at a much earlier stage of development and are unlikely to cross the TRL 7 threshold for decades.[30]

Proposals for international control of unmanned and robotic weapons systems have so far been confined to non-governmental organizations. In 2009, the International Committee for Robot Arms Control issued a statement calling for a treaty banning autonomous weapons, the equipping of robotic weapons with nuclear arms, and the deployment of armed robots in space, as well as

[27] ibid.

[28] Robotic weapons are often divided into three categories based on the amount of human involvement in their actions: human-*in*-the loop robots can select targets and deliver force only with a human command; human-*on*- the-loop robots can select targets and deliver force under the oversight of a human operator who can override the robot's actions; human-*out-of*-the-loop robots are capable of selecting targets and delivering force without any human input or interaction. "Losing Humanity: The Case Against Killer Robots," Human Rights Watch, November 2012, p. 2.

[29] Roberts, Kristin. "When the Whole World Has Drones," *National Journal*, March 22, 2013 (http://www.nationaljournal.com, accessed May 1, 2013).

[30] Swarm robotics is a research area in which large numbers of small, simple robots (a swarm or collective) are designed to work cooperatively and mimic the emergent behavior of "swarm intelligence" as exhibited by some insects. When mature, a technology of swarm robotics would have profound national security implications. However, swarm robotics is presently in the early stages of development, somewhere between TRL 2 and 3.

national reporting on the treaty's implementation.[31] In 2012, Human Rights Watch called for all states to "prohibit the development, production, and use of fully-autonomous weapons through an international legally binding instrument."[32] In April 2013, the UN General Assembly voted to adopt the Arms Trade Treaty (ATT), which aims to constrain the flow of conventional weapons to states and organizations that threaten peace and security or engage in gross violations of human rights and humanitarian law. Advocacy groups had been pushing for the inclusion of language in the ATT that would cover emerging weapons technologies, but no specific provisions pertaining to unmanned weapons appear in the final agreement.

**INFORMATION TECHNOLOGY.** Information technology (IT) is a mature but still evolving general-purpose technology which has become ubiquitous over the last three decades and has thereby created an entirely new type of national security risk: an attack against a state that uses the state's own IT infrastructure. Attacks that utilize IT infrastructure (cyber attacks or cyber warfare) differ from kinetic attacks in that attribution is difficult and attribution with absolute certainty is impossible, thus making both deterrence and retaliation problematic. Cyber attacks have the potential to return a great deal of information and/or create broad disruption, particularly in the realm of military C3I (command, control, communication and intelligence) for a small initial investment, all while allowing the perpetrator to remain anonymous. In a cyber conflict, states with the most advanced IT infrastructures are generally more vulnerable while states with little IT infrastructure are much less susceptible to disruption. Examples of how information technology can be used for offensive military purposes include the Stuxnet attack on Iran[33] in 2010 and the Israeli Air Force bombing of the nuclear site under construction at Dayr az-Zawr in Syria in 2007.[34]

National security implications of information technology and cyber conflict include:

- attacks designed to disable or destroy a state's domestic critical infrastructure (e.g., electricity grid, power plants, dams) as well as military assets and infrastructure
- disruption of defenses prior to kinetic attack (e.g., an attack to disrupt military communications prior to a conventional attack)
- placement of "logic bombs" in the critical systems of a targeted state which may then be deterred from taking certain actions through their threatened activation

For the United States, much of the difficulty in defending against cyber attack stems from the fact that the bulk of its IT infrastructure is privately owned, thus limiting the ability of

[31] Mission statement of the International Committee on Robot Arms Control (http://icrac.net/statements/), accessed May 1, 2013.

[32] "Losing Humanity: The Case Against Killer Robots," Human Rights Watch, November 2012, p. 5

[33] The Stuxnet virus targeted the control systems of Siemens industrial equipment and disproportionately infected systems that were located in Iran. Stuxnet is widely assumed to have been developed by the United States and Israel as a means of delaying Iran's acquisition of nuclear weapons by disabling the centrifuges at Iran's Natanz uranium enrichment facility.

[34] The Israeli government, apparently having purchased access to software "backdoors" in a Syrian radar system, allegedly disabled the system that would have alerted Syria to the incoming Israeli planes. On September 6, 2007, the Israeli force slipped past Syrian air defenses, bombed the target, and exited without further incident (see Mattick *et. al.*, p. 13)

government to secure it. The U.S. government is legally constrained in its ability to help companies protect their networks, in part because of privacy issues surrounding the sharing of information between the government and the private sector.[35] While industry has assumed the responsibility to protect their systems from infiltration, in many cases the financial losses stemming from information security lapses are less than the cost of the security required to prevent such losses. There may be a "public good" aspect of cyber security in that the investment to protect electronic assets, though it would benefit everyone, is under-incentivized for individuals and corporations.[36]

The idea of an international agreement governing the use of information technology is receiving attention in multiple fora, including the United Nations, the North Atlantic Treaty Organization (NATO), and the International Telecommunications Union. The Russian Federation has proposed a binding United Nations treaty on information security that would classify "information warfare" as a crime against international peace and security. Many Western countries, including the United States, have opposed the treaty, fearing that such an agreement would endorse the concept of a governmental role in controlling expression online and would be used by authoritarian governments to repress their citizens. The United States contends that the law of armed conflict, which requires the use of proportional force and the minimization of harm against civilians, applies in cyberspace, a position that is accepted by Russia but not by China.[37]

## SAFEGUARDING EMERGING TECHNOLOGIES

To what extent could a safeguards-like system of controls be established to reduce the national security threat posed by these technologies? Creating a system similar to IAEA safeguards for the mitigation of threats from emerging technologies requires solving two problems, one technical and the other political. The technical problem stems from the need to devise new safeguards tools that are effective for each technology. The political problem arises from the need to create political will among states for the adoption of a regime of controls that would constrain the use of these technologies.

**TECHNICAL CHALLENGE.** To what extent do the emerging technologies of the early 21st century resemble the technology that gave us the IAEA and the international safeguards systems? Figure 5 provides a thumbnail sketch of each of the emerging technologies considered here and the degree to which they exhibit the characteristics that made controlled nuclear fission amenable to safeguards. Controlled nuclear fission possesses all of the desired technology attributes for safeguardability and therefore serves as our benchmark. Nuclear technology's high acquisition cost, narrow range of applications, high skill barrier to utilization, requirement for a large and hard-to-conceal physical plant, dependence on a single critical component, concentration of control within a single state and staggering demonstration effect collectively made, and continue to make, safeguards possible. If fissile material were inexpensive and easy to obtain, its

[35] "Several Nations Trying to Penetrate U.S. Cyber-Networks, Says ex-FBI Official," *The Washington Post*, April 18, 2012

[36] Mattick, *et. al.*, p. 11.

[37] "In U.S.-Russia Deal, Nuclear Communication System May Be Used for Cybersecurity," *Washington Post*, April 26, 2012.

| Characteristic | Controlled Nuclear Fission | Molecular Manufacturing | Synthetic Biology | Robotics | Information Technology |
|---|---|---|---|---|---|
| Narrow range of applications? | **Yes** | No | No | No | No |
| Dependent on critical material that is difficult and expensive to obtain? | **Yes** | No | No | No | No |
| Requires large, hard to conceal physical plant? | **Yes** | No | No | No | No |
| High complexity barrier to using applications? | **Yes** | Yes (initially) | Yes | No | No |
| Exhibits physical property that can be detected remotely and measured? | **Yes** | No | No | No | No |
| Control highly concentrated at time of emergence? | **Yes** | Probably | No | No | No |
| Demonstration effect? | **Yes** | No | No | No | Yes |

***Figure 5.*** *Characteristics of nuclear technology vs. emerging technologies*

acquisition hard to detect and the complexity barriers to using its applications low enough to be surmounted by non-specialists, nuclear proliferation would have proceeded much more rapidly and to many more entities, including non-state and sub-state groups and potentially even to private individuals. The nuclear nonproliferation regime might then more closely resemble the chemical weapons control regime, with verified destruction of existing stockpiles of nuclear weapons as a principle feature.

Unfortunately, the emerging technologies of the early 21st century for the most part bear little resemblance to controlled nuclear fission. Molecular manufacturing will be a general-purpose technology and therefore inherently difficult to subject to multilateral restrictions: its applications will be too numerous and too useful for states to want to limit their utilization of it. However, it is likely that when molecular manufacturing reaches TRL 7 its control will be concentrated within a small number of states that have invested the most resources into its development. To the extent that this occurs, these states could agree among themselves to subject molecular manufacturing to a system of access controls similar to IAEA safeguards, in which access to technology is made conditional upon acceptance of verified limits on its use. In contrast to molecular manufacturing, the toolkit of synthetic biology is already widely commercially available in many countries and on the Internet, making the task of safeguarding it much more challenging. Robotics is more widely disseminated still, and information technology has spread worldwide. The challenge of devising a safeguards system for these technologies and getting broad international acceptance for them would be enormous.

**Adaptation of IAEA Safeguards to Emerging Technologies.** Setting aside for the moment the question of whether and how an international agreement could be achieved and a corresponding verification authority established, to what extent could the principles and techniques used by the IAEA in carrying out its nuclear safeguards mission be applied productively to these technologies? Obviously, those safeguards techniques which rely on measurements of radiation will not be applicable beyond nuclear technology. However, many other IAEA safeguards principles and techniques could be transferred more or less directly from nuclear technology to emerging technologies with good effect. Figure 6 provides an overview of the potential applicability of various existing safeguards techniques to the emerging technologies that are

considered here. Possible applications of (or adaptations of) established IAEA international safeguards techniques to the emerging technologies that have been considered here are outlined below.

***Molecular Manufacturing***. Many IAEA safeguards techniques could be adapted for the verification of an international agreement governing use of nanotechnology and molecular manufacturing:

- state self-reporting via declared facilities (this would likely require the development of a regulatory infrastructure within each participating state to register and account for desktop nanofactories owned by private entities and individuals)
- material sampling to verify adherence to treaty-based restrictions on size or type of nano-material (particle, machine, etc.)
- weight and volume measurements to verify restrictions on quantities of nanomaterials that may be produced in a given facility or during a specified period of time
- on-site monitoring via a permanent inspector presence at facilities in which nanotechnology or molecular manufacturing applications are used
- unattended and remote monitoring via sensors (desktop nanofactories could come pre-equipped with sensors that report to the international verification authority the type and quantity of items manufactured and which prohibit the manufacture of restricted items)
- containment, surveillance and physical protection to prevent unauthorized access to and use of nanomanufacturing devices
- environmental sampling to detect the presence of nanoparticles
- open-source information analysis to detect evidence of prohibited uses of molecular manufacturing

***Synthetic biology.*** As with nanotechnology, an international agreement governing the uses of synthetic biology could make use of adapted versions of many IAEA safeguards techniques:

- state self-reporting via declared facilities (as with molecular manufacturing, a state-level regulatory mechanism would be needed to ensure compliance by private entities with reporting requirements)
- material sampling to verify adherence to treaty-based restrictions on types of synthetic organisms
- weight and volume measurements to verify restrictions on quantities of bioengineered substances, materials or organisms that may be produced in a given facility or during a specified period of time
- on-site monitoring via a permanent inspector presence at facilities where bioengineered or synthetic organisms are created and used
- unattended and remote monitoring via sensors designed to detect specific engineered substances and organisms (this may require creation of an international registry of synthetic organisms)

| Safeguards Principle | IAEA Safeguards Technique | Molecular Manufacturing | Synthetic Biology | Robotics | Info Tech. |
|---|---|---|---|---|---|
| **State Self-Reporting** | Declared facilities | ● | ● | ● | |
| **Non-Destructive Analysis** | Radiation detection techniques (*γ*, neutron) | | | | |
| **Destructive Analysis** | Material sampling | ● | ● | | |
| **Physical Property Measurement** | Design information verification (ground penetrating radar, 3D laser range finder, ultrasonic thickness gauge) | | | | |
| | Volume measurement (portable pressure measurement device) | ● | ● | | |
| | Weight measurement (load cell based weighting system) | ● | ● | | |
| **On-Site Monitoring** | Permanent inspector presence | ● | ● | | |
| **Unattended and Remote Monitoring** | Weight, volume, temperature, flow monitoring | ● | ● | | |
| | GPS tracking | ● | ● | ● | |
| | Video surveillance | ● | ● | | |
| | Unattended monitoring systems (including sensors for radiation, temperature, pressure, flow, vibration, optical and electromagnetic) | ● | ● | ● | |
| **Containment** | Single use seals (metallic, adhesive) | ● | ● | | |
| | In situ verifiable seals (fiber optic, ultrasonic, electronic, radiofrequency) | ● | ● | | |
| | Container verification (laser mapping, laser item identification system) | ● | ● | | |
| **Surveillance** | Optical surveillance (cameras, CCTV) | ● | ● | | |
| **Environmental Sampling** | Bulk analysis (mass spectrometry, scanning electron microscopy) | ● | ● | | |
| | Particle analysis (swipes, air sampling) | ● | ● | | |
| **Material Flow Measurement** | Process holdup measurement | | | | |
| **Physical Protection** | Access control | ● | ● | | |
| | Intrusion detection | ● | ● | | |
| **Open-Source Information Analysis** | Patent applications | ● | ● | ● | |
| | Published scientific and technical literature | ● | ● | ● | |
| | Publicly available information (news media, NGOs, governments) | ● | ● | ● | |
| | Commercial satellite imagery | ● | ● | ● | |

***Figure 6.*** Applicability of Existing IAEA Safeguards Techniques to Emerging Technologies

- containment, surveillance and physical protection to prevent unauthorized access to biotech facilities
- environmental sampling to detect the presence of bioengineered organisms
- open-source information analysis to detect evidence of prohibited uses of synthetic biology

***Robotics.*** Fewer safeguards techniques are applicable to a verification regime for unmanned and robotic weapons. Some, however, might have value if properly adapted:

- state self-reporting of activities involving unmanned or robotic weapons systems (a modification of declared facilities)
- unattended and remote monitoring of unmanned and robotic weapons to detect prohibited modifications and uses (for example, a "black box" that records the date, time and duration of each use similar to that incorporated into Tasers. Such a device built into an unmanned or robotic weapon system could not only record information about its use but also transmit information about the system's location and status to the international verification authority. Also, GPS tracking of specified unmanned or robotic systems could be used to verify compliance with treaty-based limitations on where such systems can be deployed.)
- open-source information analysis to detect evidence of prohibited activities (e.g., patent applications or software for prohibited classes of weapons)

***Information technology.*** There appear to be no existing safeguards techniques that would be readily transferrable to verifying compliance with an international agreement on the rules of cyber warfare or the appropriate military uses of information technology. There are no materials to be measured or sampled, no items to be accounted for, and nothing to be inspected. We shall therefore need to look elsewhere for IT safeguards techniques, assuming any exist at all.

Emerging technologies themselves may open new possibilities for safeguards and verification techniques that can then be applied to any prospective regime for their control, or even to existing technology control regimes. Nanotechnology in particular offers the prospect of new tools to effectively and profoundly strengthen the nonproliferation regimes for chemical and biological weapons. Sensitive, selective and inexpensive nanotech sensors and materials could detect and bind components of chemical, biological or radiological weapons on the atomic or molecular level, due to the large surface-to-volume ratio of nanoparticles or of nanoporous material.[38] The combination of nanotechnology and robotics would seem to lend itself well to verification applications: one can envision nanoscale autonomous sensors and surveillance devices being applied for this purpose, although as previously noted, the same technology could promote crisis instability when used by governments for covert surveillance.

**POLITICAL CHALLENGE.** Emerging technologies take time to develop, and this time can be used by the international community to prepare for the disruptions that the mature technologies will engender. Ideally, by the time an emerging technology has reached TRL 7, the international community should have: 1) identified and understood the risks; 2) made appropriate policy; 3) designed the necessary institutions; and 4) established these institutions, both domestically and internationally.[39] Unfortunately, this sort of advance planning is very difficult for governments to achieve in practice. Political will is finite, and governments must direct their limited supply toward addressing today's problems which usually means there is little left over to address problems that may not manifest for decades. Creating a risk-mitigation regime for an emerging technology similar to the one that has grown up around controlled nuclear fission would require

[38] Ibrugger, Lothar (rapporteur), "The Security Implications of Nanotechnology," Report of the NATO Parliamentary Assembly (179 STCMT 05 E), 2005 (http://www.nato-pa.int?Default.asp?SHORTCUT=667, retrieved December 21, 2012)

[39] "Administration Options for Molecular Manufacturing," Center for Responsible Nanotechnology, op. cit.

an underlying international legal instrument with sufficient – and sufficiently important – state subscribers to make it matter; a normative infrastructure that develops from the agreement; an internationally-administered control authority with broad legitimacy; corresponding regulatory infrastructures in the participating states; and, in some cases, a corresponding export control regime. What sorts of conditions or incentives could create the political will that would be needed to lift such a heavy payload?

The desire for transparency surrounding states' uses of an emerging technology could conceivably create the political will necessary to establish an international agreement governing that technology. So could the desire to cement a technological advantage for a particular state or group of states, although this desire would presumably be met by a countervailing desire on the part of technology laggards to impose no constraints on the acquisition or uses of the technology, at least until they have caught up. However, as with controlled nuclear fission, the most powerful generator of political impetus for a control regime on a given technology would probably be a compelling demonstration effect in the form of a highly-publicized weaponization of or accident involving a new technology, although this is not the sort of thing that governments can count on (nor can they seek to bring it about). Given these limitations, the prospects for achieving an international control regime for each of the emerging technologies considered here are sketched briefly below.

**Molecular manufacturing.** Although no government has yet offered a proposal for an international agreement restricting nanotechnology research or applications, academic and policy groups have given some though to how the international community might best manage the national security consequences of a mature technology for molecular manufacturing. The Center for Molecular Nanotechnology (CMN) suggests that careful administration of molecular manufacturing technology will be required in order to mitigate its inherent danger. Like controlled nuclear fission, molecular nanotechnology has features that simultaneously increase the technology's risk while facilitating the sort of restrictions on its use that could become the components of an international administration program. For example, the compactness of molecular nanotechnology could make possible the safeguarding of a human-scale nanomanufacturing product, such as a personal nanofactory, by incorporating dedicated security or monitoring hardware directly into the device. Experts have also suggested that both the Chemical Weapons Convention and the BTWC be supplemented with appropriate "clarifying interpretation that nanotech-enabled microscopic systems that can enter the body, damage life processes and are partly or fully artificial are included" within the scope of each agreement.[40]

However, CMN concludes that safe use of molecular nanotechnology will ultimately require "the creation of a single, trustworthy international administration to impose tight controls on the technology," but that unless the technology "is made widely available for a broad range of applications, there will be strong incentives for states to pursue their own independent molecular nanotechnology programs."[41] This reading of the impact of molecular

[40] "The Security Implications of Nanotechnology," op. cit., p. 6

[41] "Administration Options for Molecular Manufacturing," Center for Responsible Nanotechnology (http://www.crnano.org/administration.htm), accessed April 13, 2013.

nanotechnology and the resulting prescription for mitigating its risk tracks very closely with the evolution of controls on nuclear technology and the creation of the IAEA. The CMN proposal even goes so far as to call for "a closely-guarded crash program to develop a self-contained, secure molecular manufacturing system" which would then be made available only for the manufacture of approved products or classes of products.[42] This proposal would develop molecular manufacturing to maturity in a manner that precisely duplicates the development of controlled nuclear fission, right down to its own Manhattan Project. Thus nanotechnology, despite being a GPT, could be a plausible candidate for a treaty-based international control regime and accompanying system of safeguards administered by an authority analogous to the IAEA, assuming that the technology is developed to maturity in a manner that mirrors the development of controlled nuclear fission. If not, attempting to establish international control over molecular manufacturing after it has already worked its changes on society will likely prove futile.

**Synthetic biology.** Application of synthetic biology that are expressly conceived as weapons, such as artificial organisms incorporating enhanced virulence factors, are already prohibited by the BWTC, although verification remains problematic. Otherwise, synthetic biology is very tightly coupled with proprietary commercial development and thus difficult for governments to regulate. In March 2012, over one hundred environmental and civil society groups issued a collective statement on "The Principles for the Oversight of Synthetic Biology which calls for a worldwide moratorium on the release and commercial use of synthetic organisms until more robust regulations and rigorous biosafety measures are established. The group also called for an outright ban on the use of synthetic biology on the human genome or human microbiome.[43] A 2007 study concluded that "with very few exceptions, synthetic genomics would not now be the technology of choice for a bioterrorist or a nation-state hoping to develop a virus for use as a weapon. Within five to ten years, however, it may very well be the case that synthesis will be easier than other means of obtaining a virus."[44] The creation of an international control regime for synthetic biology would likely be hindered by the same factors that contribute to the continuing absence of a verification regime for the BWTC, compounded by the intensely commercial nature of most synthetic biology development.

**Robotics.** Despite the popular, media and academic attention being given to unmanned and robotic weapons systems and the calls for an international agreement banning certain kinds of lethal autonomous systems, it seems unlikely that an international agreement governing robotic systems will be forthcoming anytime soon. UAVs are already widespread and have far too much utility, military and otherwise, for states to be willing to restrict their use. It may be possible to reach an agreement to ban entirely autonomous systems capable of exercising

[42] ibid.

[43] Boyle, Alan (March 14, 2012). "What To Do about Synthetic Life?" http://cosmiclog.msnbc.msn.com/_news/2012/03/13/10672301-what-do-do-about-synthetic-life, retrieved May 3, 2013.

[44] Garfinkel, Michele S., Drew Endy, Gerald L. Epstein and Robert M. Friedman. "Chapter 35: Synthetic Biology," *The Hastings Center Bioethics Briefing Book*, p. 164. http://www.thehastingscenter.org/uploadedFiles/Publications/Briefing_Book/synthetic%20biology%20chapter.pdf, accessed May 3, 2013.

lethal force without a "human-in-the-loop," since no states have openly deployed such systems yet. But verification of such a ban would be extremely difficult, even if sufficient political will could be mustered for a verification system that is highly intrusive. The difference between a human-controlled or human-mediated robotic weapon system and a fully-autonomous lethal robotic weapon system is simply a matter of software. It would be extremely easy to re-program treaty-compliant weapon systems to function as a prohibited type of system, making the verification task functionally impossible for even the most robust inspectorate. Perhaps the best chance at an agreement banning an entire category of robotic weapons would be a ban on equipping robotic systems with nuclear weapons, since such a system would seem to have little military utility and many potential drawbacks.

**Information technology.** Of the technologies considered here, information technology is the only one that is presently the subject of an ongoing international negotiation as described previously. As of now, there is only one international agreement in place that governs use of the Internet: the Council of Europe's Cybercrime Convention, adopted in 2001. However, the Cybercrime Convention is widely viewed as unsuccessful due to failures of verification and enforcement.[45] States cannot determine quickly or easily when their IT systems are being attacked, and once the attack is discovered, the computer or geographic source of the attack often cannot be ascertained quickly or precisely. If a computer or geographic source is identified, it is hard to know whether the responsibility for the attack lies there or with a different computer somewhere else. Even if a state has certain knowledge about which computer was the ultimate source of the attack, it is hard to know whether the human agent behind it is a private individual or a government. If the latter, it is frequently hard to determine the state affiliation.[46] Any conceivably effective verification regime for an international agreement on the use of information technology would require such extensive government monitoring of the Internet as to prove unworkable in many countries. For these reasons, it seems unlikely that an effective control regime for the military use of information technology will emerge in anything that closely resembles today's international environment.

**ROLE OF THE IAEA.** The IAEA is the obvious model for any future international control authority charged with promoting the peaceful use of an emerging technology while verifying the absence of any prohibited uses under the terms of such agreements as may be forthcoming. To what extent could we -- or should we -- leverage the IAEA itself for this purpose? The IAEA enjoys a considerable advantage compared to other potential implementing mechanisms for any new control regimes that would govern the use of emerging technologies. These include its established reputation for independence and objectivity, its international character and ability to build consensus worldwide (and its considerable experience in doing so), and its capability to establish – and assist member states in complying with – international norms and standards governing a dual-use technology.[47] In addition, the IAEA is responsible for ensuring that the advantages of the technology under its purview are used to benefit human well-being and sustain

[45] Goldsmith, Jack "Cybersecurity Treaties: A Skeptical View (February 2011)," in *Future Challenges in National Security and Law*, edited by Peter Berkowitz, http://www.futurechallengesessays.com, pp. 1-2

[46] ibid, pp. 3-4

[47] "*20/20 Vision for the Future: Background Report by the Director General for the Commission of Eminent Persons."* International Atomic Energy Agency, May 23, 2008 (GOV/2008/22-GC (52)/INF/4 Annex), p. 4.

socioeconomic development while also seeking to ensure that the risks associated with nuclear technology are minimized.[48] Finally, the IAEA has a role in ensuring the safe and secure application of nuclear technology and in reducing the likelihood of accidents through human or technical failure. These are all desirable characteristics for any future control regime governing access to and use of molecular manufacturing, synthetic biology, information technology, and/or robotics. By far the easiest route to creating a verification authority for any future agreements governing these technologies would be the adaptation of the IAEA itself.

The obvious disadvantage of attempting to use the IAEA as a direct platform for launching international verification and control systems for emerging technologies is that the IAEA's current expertise is confined entirely to nuclear technology. Any new expertise would need to be "bolted on," possibly compromising the Agency's effectiveness in executing its existing nuclear safeguards mandate. This danger is especially acute if governments attempted to leverage the IAEA for an emerging technology verification mission without a corresponding increase in Agency resources. The best approach to capitalizing on the success of the IAEA and the international safeguards system for a new technology safeguards mission would be the establishment of a dedicated verification authority for the relevant international agreement and the subsequent incorporation the IAEA's lessons learned from fifty years of safeguards implementation. This could be accomplished by establishing short-term dedicated working groups comprised of personnel from the IAEA and the new verification authority. These working groups could provide guidance to those governments and international organizations that are working to set up the new verification entity. Once the new verification authority is established, IAEA expertise could be further leveraged through longer-term detail assignments and personnel exchanges.

## CONCLUSIONS

As the 21st century progresses, the national security challenge posed by emerging and converging technologies will become ever more acute. But the continued success of the nuclear nonproliferation regime and of the IAEA safeguards system demonstrates that successful management of the national security risks associated with emerging technologies is not only achievable but is sustainable over the long term, even in the face of serious tests. The ongoing evolution of IAEA safeguards from a system that verifies legitimate and declared uses of nuclear technology into a system for discovering illicit and clandestine ones shows that safeguards are capable of adapting successfully to changing political and technological requirements. In the coming decades, the IAEA and the international safeguards system will remain vital instruments of global security, not only in their role as guarantors of the nuclear nonproliferation regime but also increasingly as the pathfinders to a secure human future that is protected, as much as it ever can be, from the risks that come with our ever-evolving technology.

[48] ibid, p. 12

# Approaching acquisition path analysis formally - a comparison between AP and non-AP States

Clemens Listner[1], Morton J. Canty[1], Arnold Rezniczek[2], Gotthard Stein[3] and Irmgard Niemeyer[1]

[1]Forschungszentrum Jülich, Germany
[2]UBA GmbH, Herzogenrath, Germany
[3]Consultant, Bonn, Germany

**Abstract**

In the past, the IAEA has planned its activities mainly based on the presence of nuclear material. However, resources should be spent where they are needed most. Therefore, a new risk model was developed to change the inspection system to a comprehensive, objective-driven approach where the State is considered as a whole, the so called State-level concept (SLC). Acquisition path analysis (APA) is a key element of the State-level concept. By considering the State's nuclear profile, the APA generates a list of acquisition paths ranked by their attractiveness for the State. Currently, this process is mainly based on expert judgment. However, the IAEA's requirements state that APA must be objective, reproducible, transparent, standardized, documented and as a result non-discriminatory.

A formal approach fulfilling the requirements was set up by the authors in the past [1]. This methodology is based on a three step approach. The process starts in the first step with the parametrization of the network. In the second step, the network is analyzed in order find all acquisition paths for a State. Finally, game theory is used in the third step to model the decisions made by the IAEA and the State.

In this paper, an advanced methodology will be presented. Improvements were made in the interface definition between the three stages. Also, the general network model was updated and the automatic visualization of acquisition paths was accomplished. Furthermore, a prototype implementation will be shown.

The advanced methodology was applied to two test non-nuclear weapon States under comprehensive safeguards agreements with the IAEA. Both States hold complex fuel cycles with only small technical differences. However , only one State is supposed to have the additional protocol (AP) in force. The example will show how the presence of the AP influences the detection probabilities of illegal behavior. As a consequence, these examples also indicate where to best focus safeguards efforts.

# 1 Introduction

Due to the experiences made in the past, the IAEA has developed a vision of a new verification model - the State-level concept (SLC). While the former approach has focused on declared nuclear material and facilities, the new concept concentrates on facts about a State. In order to increase effectiveness and efficiency, the IAEA wishes to migrate from a mechanistic verification procedure to a risk-based prioritization of its activities.
While differentiating between States due to different risk levels seems reasonable, one has to assure that no single State will be discriminated. Therefore, the State-level concept is to be objective, transparent, reproducible, standardized and documented (for details see Listner, Canty, Rezniczek, Stein, and Niemeyer [1]). Furthermore, the State-level concept should be applicable to all States with commitments regarding nuclear non-proliferation, i.e. item- or facility-specific safeguards agreements (INFCIRC/66), comprehensive safeguards agreements (CSA) with and without additional protocol (AP) as well as voluntary offer agreements (VOA) in nuclear weapon States (NWS).
According to Cooley [2], the State-level concept comprises three steps leading to a specific State-level safeguards approach:

1. Identification of plausible acquisition paths.

2. Specification and prioritization of State-specific technical objectives.

3. Identification of safeguards measures to address the technical objectives.

In the present paper, we will concentrate on the first step of the process, the acquisition path analysis (APA), as this is still a major difficulty when elaborating a State-level approach.
According to IAEA [3], an acquisition path is defined as a sequence of activities which a State could consider in order to acquire weapons usable material. The APA analyzes all plausible acquisition paths, aiming to determine whether a proposed set of safeguards measures will be effective.
Up until now, the IAEA has implemented APA mainly based on expert judgment. This has led to a procedure that, although standardized, cannot fulfill the requirements mentioned above. Therefore, the IAEA is looking for a methodology and software tool that helps structuring the process of APA. The tool should provide for visualization of the acquisition paths in order to help the analyst maintain an overview of the situation in a State. It should automate the process in order to be independent of subjective reasoning and thus guarantee reproducibility. Finally, a software tool assisting the analyst should integrate into existing systems and models at the IAEA.
In the following sections, we describe a new formalized procedure using network analysis techniques and game theory in order to assess proliferation risks and help distributing the inspectorate's resources in a reasonable way. In order to demonstrate the method, two case studies were carried out. Both cases were based on States with advanced fuel cycles. While the technical capabilities only slightly differ, they hold different safeguards agreements with the IAEA. The first State has a CSA as well as an AP based on INFCIRC/540 (corrected) including a broader conclusion in force. The second State only holds a CSA with the IAEA.

# 2 Materials and Methods

As mentioned in the previous section, the IAEA's physical model serves as a basis for APA. It comprises an overview of all known relevant processes for converting nuclear source material to weapon usable material (see definition in IAEA [3]). In Figure 1, a generic version is depicted. The yellow boxes represent material forms which are transformed to other material forms by using specific processes symbolized by arrows in the model. In the figure, the acronym "DU" stands for direct use, "IU" for indirect use and "NU" for natural uranium. Within the physical model, an acquisition path always starts at the "Origin" box followed by the diversion or undeclared import of the first material form. From there, consecutive process steps consisting either of misuse of declared facilities or of processing in undeclared facilities are needed in order to finally acquire weapons usable material, i.e. any material in the physical model with the prefix "DU".

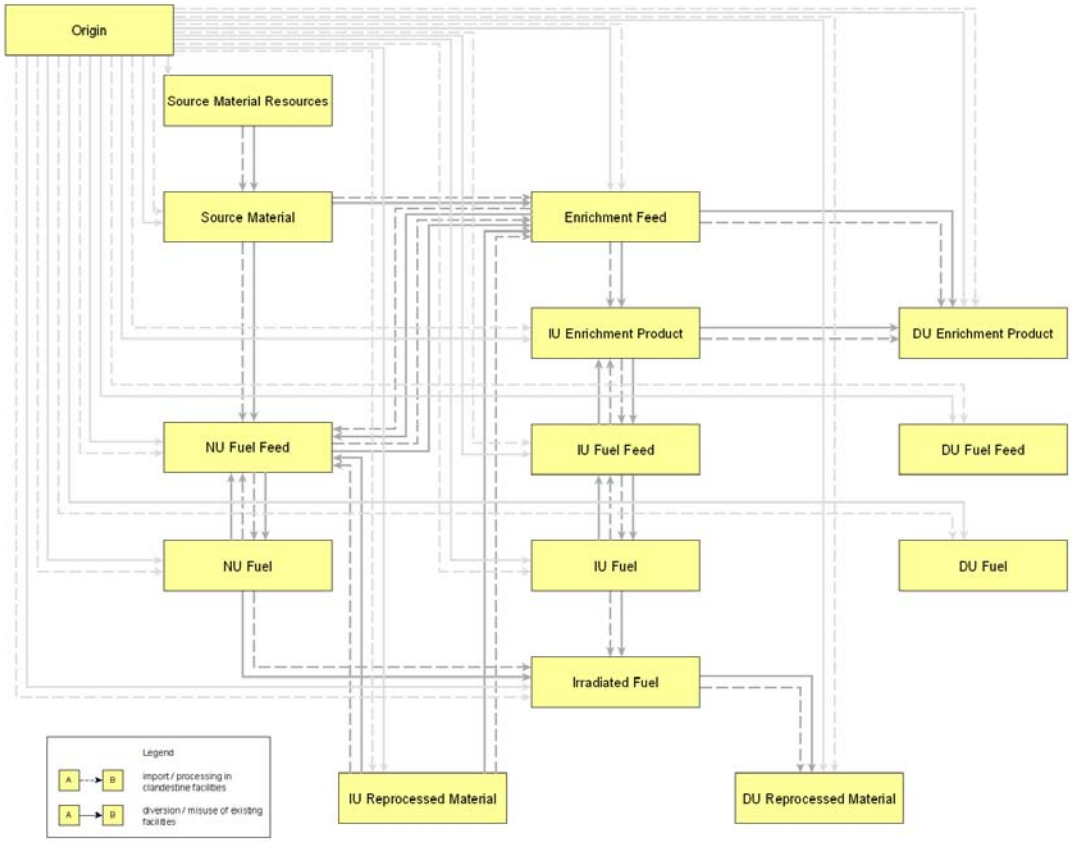



Figure 1. Generic physical model.

From a mathematical point of view, the physical model can be considered as a directed graph. Therefore, it is possible to apply graph theory to the APA problem in order to find all technically plausible paths.
In a directed graph, nodes can be connected by edges. To each edge a scalar number can be associated to measure the edge's length. Sequences of edges are called paths whose lengths are calculated by the sum of the edge weights. The concept of graph theory applied to acquisition path analysis implies a node being a material form in the physical model, an edge representing a process and a path standing for the acquisition path itself. Edge weights can be used to reflect the attractiveness of a process.

Using the model of a mathematical graph, acquisition paths can be found using several algorithms. The shortest path between two nodes can be found using e.g. the Dijkstra algorithm (see Dijkstra [4]). However, for a comprehensive analysis of a State's nuclear options, all paths from "Origin" to any of the "DU" material forms have to be assessed. This can be easily accomplished using enumeration techniques that prevent cycles contributing to an acquisition path.
Besides finding acquisition paths, APA ought to assess paths with respect to their suitability for a nuclear weapons program. As this comprises a strategic aspect, game theory is the appropriate tool to accomplish this task.
Game theory is a mathematical approach with the ability to model strategic situations between opposing players. By strategic situation, a choice between different decision alternatives is meant where decision making does not only depend on the protagonists' own courses of action but also on those of their opponents. Applied to APA, the inspectorate and the State are the opposing players. The strategies of the State are the acquisition paths themselves, as well as the strategy of compliant behavior. For the inspectorate, the different safeguards approaches, i.e. the inspection of a subset of all processes, can be considered as strategies. Each combination of the players' strategies may be associated with the players' utilities, thus leading to a bi-matrix representation of the game (for details see Canty [5]).
Using this problem formulation, game theory provides a solution using the concept of Nash equilibrium. A Nash equilibrium in a two-person game is defined as a strategy combination for which neither of the players can deviate unilaterally in order to increase his or her utility. Thus, a relative scale is defined upon which different acquisition path configurations can be evaluated and compared. The effectiveness of a given inspection regime can be measured on the basis of the inspectorate's payoff. The most efficient strategy for the IAEA is that minimum effort strategy which is part of a Nash equilibrium in which the State behaves legally.
Using both graph theory and game theory, a new 3-step approach to acquisition path analysis was established. This approach is depicted in Figure 2.

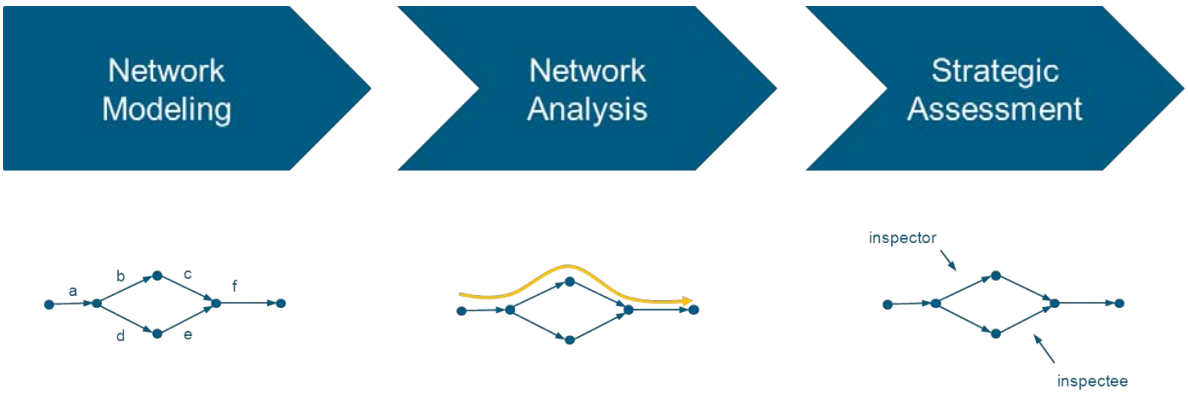


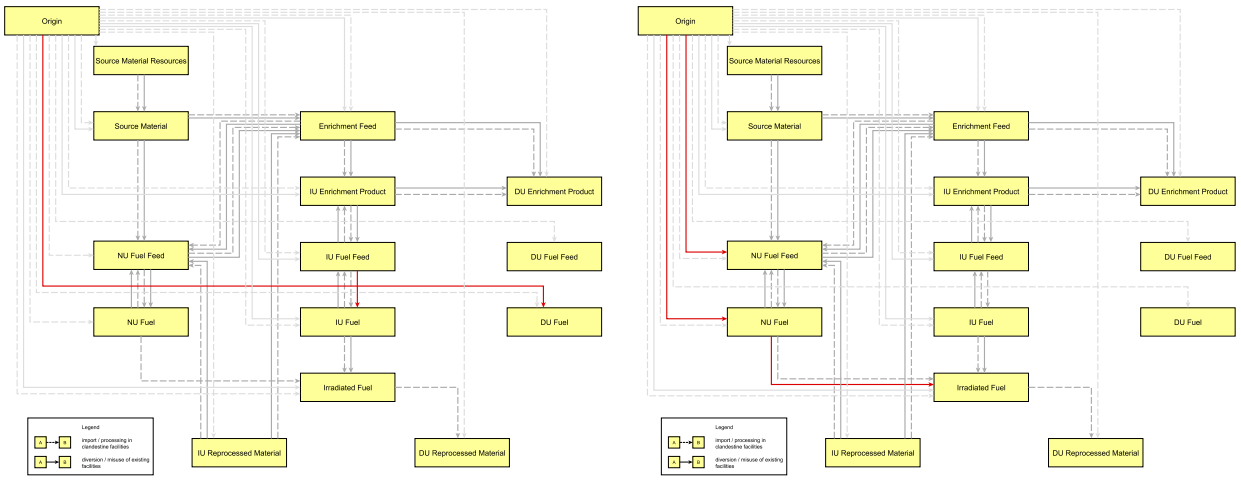

Figure 2. 3-step approach to acquisition path analysis.

The first step, network modeling, parameterizes the model. With respect to the edge weights, GIF PR/PP (see GEN IV International Forum [6]) proposes six measures for evaluating proliferation resistance which can also be used to model attractiveness in the case of APA. For this paper, three of these measures, technical difficulty (TD), proliferation time (PT) and proliferation cost (PC), are taken into account. Using these dimensions, the analyst is given the opportunity to rate each process based on a scale from 0 (very attractive), to 3 (very

unattractive).
In the second stage of the process, the network analysis, all paths with their respective lengths are enumerated and sorted in decreasing order of attractiveness. Additionally, the paths are visualized using the GraphML format (see Brandes, Eiglsperger, and Lerner [7]) and yEd Editor (freely available at www.yworks.com). This step is carried out in a fully automatic way using a Python script including the NetworkX toolbox (see Hagberg, Schult, and Swart [8]).
Finally, in the third stage of the process, the strategic assessment, the strategic options of both the IAEA and the State are evaluated using the game theoretic approach described above. To accomplish this, actions of the inspectorate are associated with costs and presumed detection probabilities. Based on a cost threshold $W$, it is possible to compute the Nash equilibria sequentially and thus to obtain the inspectorate's minimum effort strategy for inducing compliant behavior on behalf of the State.

# 3 Example Case Studies

In order to illustrate the methodology's workflow presented in Section 1, it was applied to two non-nuclear weapon States (NNWS) signatory to the non-proliferation treaty (NPT). Both States were modeled as having a CSA in force. However, for State A an additional protocol was assumed to be in force with the broader conclusion drawn. For State B no additional protocol was considered.
At first, the generic physical model in Figure 1 was adopted to the situation in the two countries leading to the removal of several edges: Both States were assumed to have no declared reprocessing, i.e. the edges representing the diversion of reprocessed material as well as the edge for the misuse of an existing reprocessing facility were deleted. Also, direct use material was only supposed to be present as direct use fuel in State A. Therefore, the edges representing diversion of those materials were removed except for direct use fuel diversion in the case of State A. Furthermore, no declared natural uranium fuel cycle was assumed in State A, thus the edges for natural uranium diversion and the misuse of a heavy water reactor were removed. On the other hand, in State B no reactor utilizing direct use fuel was modeled. Therefore, diversion of this type of fuel was not part of the physical model in the case of State B. The modified physical models are shown in Figure 3.
Both States were modeled to have a complex nuclear fuel cycle with a great deal of nuclear experience and know-how. Therefore, the attractiveness ratings were chosen similarly for State A and State B.
Regarding the inspectorate costs, the Safeguards Implementation Report (SIR) from 2011 served as a basis for estimation. For State A, the costs of all possible safeguards measures were assumed to be 5,900,000 EUR. Accordingly, for State B this amount was supposed to be 6,600,000 EUR. Based on the person-days of inspection, these costs were split and associated to particular areas of the physical model.
Expert judgment was applied to assess the attractiveness of each edge in the physical model. The IAEA's detection probability for clandestine processing was set to 95% in State A due to the presence of the AP and a broader conclusion. The figure of 95% detection probability is derived from the SIR statement saying that in a State with CSA, AP and broader conclusion "'all nuclear material remained in peaceful activities"'. As there is always a chance to be wrong, a detection probability of 95% or equivalently a non-detection probability of 5% was assumed. In case of State B without the AP and the broader conclusion, the SIR




(a) State A. (b) State B.

Figure 3. Physical models for the two cases with differences highlighted in red.

statement is restricted to declared nuclear material implying that the IAEA is unable to draw a conclusion regarding undeclared activities. But there is always a certain chance of obtaining other kinds of information which is considered to be significantly higher than the non-detection probability for clandestine processing of 5% in case of State A. Therefore, the detection probability for clandestine processing was assumed to be 20% in State B.

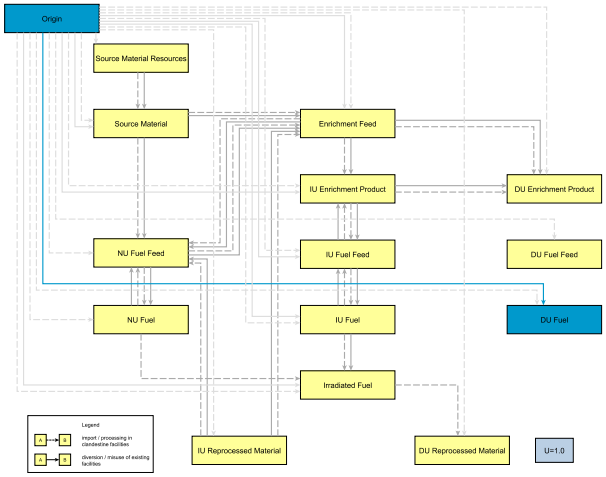
After these steps, the adjacency matrix was available for further analysis. To accomplish the modeling, Microsoft Excel was used, allowing for an export of the relevant data.

Based on the outcome of the first step, all paths between the “Origin” node and any “DU” node were enumerated, visualized and sorted according to their attractiveness in the second step. This step uses the Python script mentioned in the previous section. Thus, a list of 1041 paths for State A was generated, while 814 paths resulted for State B. Moreover, for each path a separate chart was generated visualizing the respective acquisition path and its attractiveness (see examples in Figure 4).

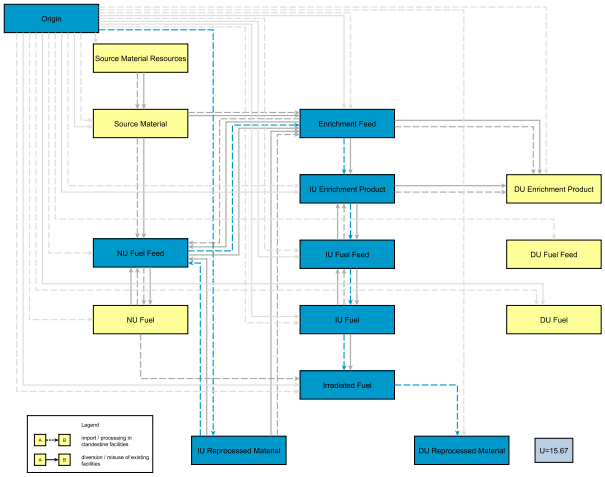
In the third step, the acquisition paths together with compliant behavior were considered as the State’s strategies. This led to a total number of 1042 strategies for State A and 815 strategies for State B. For the IAEA, any combination out of 16 distinct activities for State A and 18 distinct activities for State B were assumed to be a strategy. The costs for each combination of inspectorate activities were calculated based on the input from the SIR. By setting a limit on the inspectorate’s costs, a set of strategies for the IAEA was determined. Then, the Nash equilibrium was calculated. If it did not comprise compliant behavior on behalf of the State, the threshold was increased by the amount of 100,000 EUR until the maximum amounts of 5,900,000 EUR or 6,600,000 EUR where reached. This approach led to the results in Figure 5.

Regarding the equilibrium strategies for a maximum budget of 800,000 EUR given State A, the inspectorate player chose a mixed strategy consisting of seven particular inspection strategies. The areas of the physical model that each strategy focused on are marked by an ”‘x”’ in Table 1. The player representing the State, on the other hand, chooses compliant behavior in this situation.

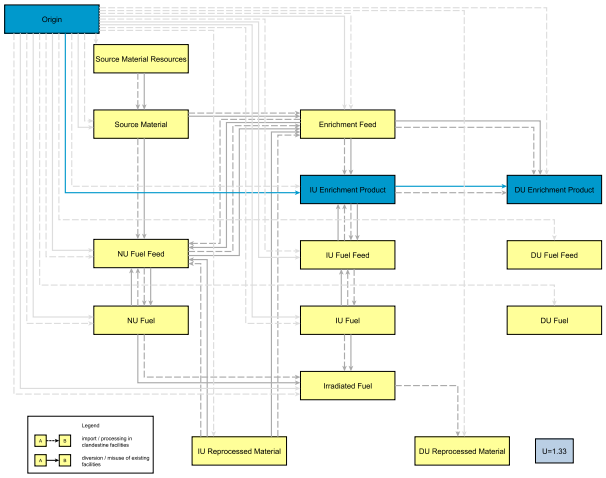
According to the model, a budget of 800,000 EUR was sufficient to induce legal behavior in the case of State A. Further increasing $W$ did not change this behavior. Following the comparison of the estimated with the actual budget of 5,700,000 EUR spent in State A, a



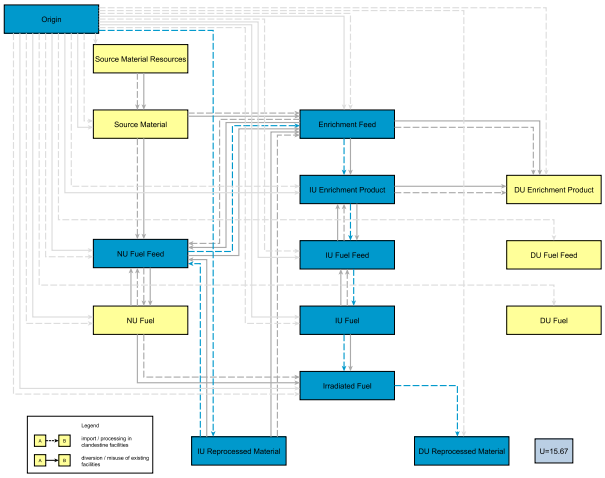

(a) Most attractive path for State A.




(b) Least attractive path for State A.




(c) Most attractive path for State B.




(d) Least attractive path for State B.

Figure 4. Two paths for each of the example cases.

| **Strategy #** | 1 | 2 | 3 | 4 | 5 | 6 | 7 |
|---|---|---|---|---|---|---|---|
| **Probability** | 2% | 5% | 24% | 32% | 21% | 6% | 10% |
| **Costs [kEUR]** | 793 | 785 | 727 | 689 | 794 | 741 | 741 |
| **DU Fuel Diversion** | x | x | x | x | x | | |
| **IU Enrichment Product Diversion** | x | x | | | | x | |
| **Enrichment Feed Diversion** | x | | | | | | x |
| **Conversion I Misuse** | x | | x | | | | |
| **Irradiated Fuel Diversion** | | x | | | | | |
| **Undeclared Import** | | | x | | | x | x |
| **Enrichment Misuse** | | | | x | | | |
| **Undeclared Processing** | | | | | x | | |

Table 1. Equilibrium strategy for the inspectorate in case of State A and $W = 800,000$.

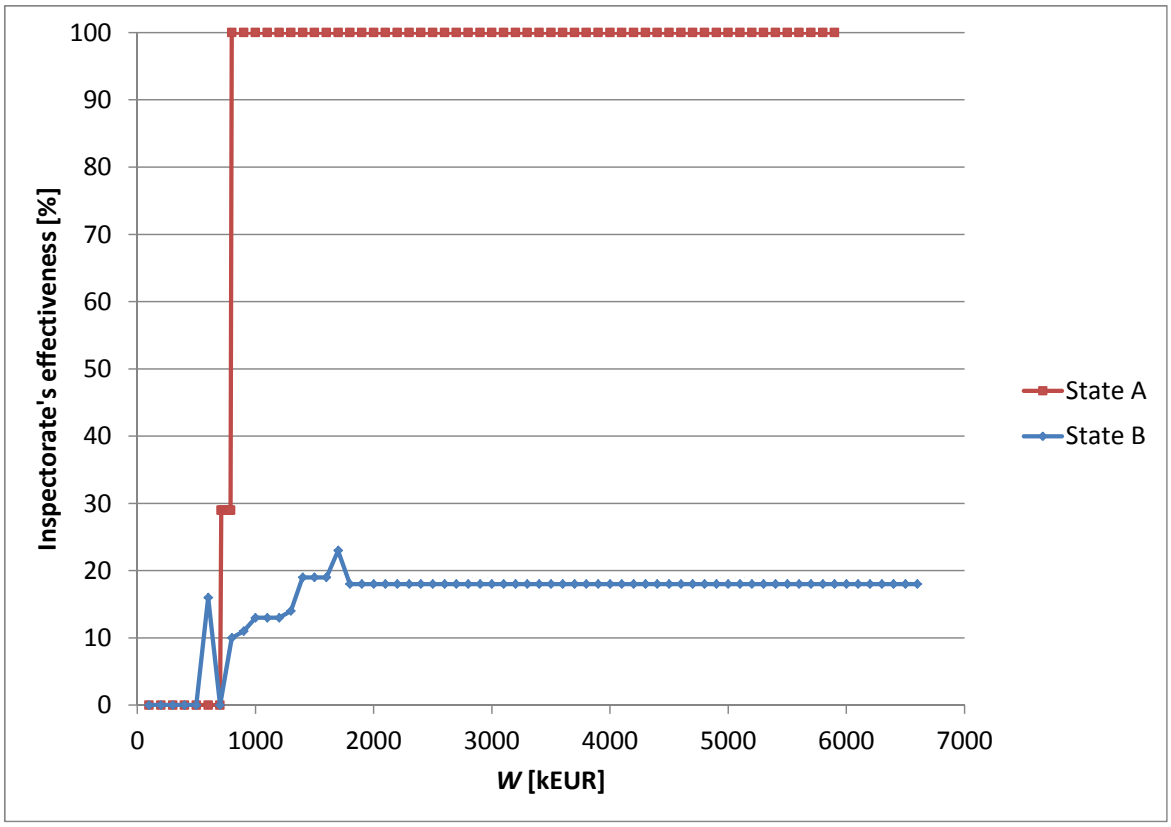



Figure 5. The inspectorate's effectiveness for the two States depending on the maximum budget.

significant increase in efficiency without losing effectiveness should be considered possible.
In the equilibrium situation for the model of State B and $W$ set to the maximum budget of 6,600,000 EUR, the inspectorate player chooses the strategy comprising all possible safeguards measures. The player representing State B, though, chooses an acquisition path leading to highly enriched uranium. This path is mainly based on processing in clandestine facilities.
This example turns out that deterrence cannot be achieved by the inspectorate in case of State B, as the detection probability for undeclared processing only amounts to 20% due to the absence of the additional protocol.
It should be noted that, in general, effectiveness values between 0 and 100% can arise when the State's Nash equilibrium strategy is illegal behavior given a finite chance of detection. This situation does not arise in the case of State A, however.

# 4 Conclusions and Outlook

Within this paper, an advanced 3-step methodology for acquisition path analysis was presented. The approach fulfils the IAEA's requirements regarding acquisition path analysis of being objective, reproducible, transparent, standardized and documented. As a result, the methodology accounts for differentiation between States without discrimination. Moreover, the procedure is modular and can be automatized. However to select the model's parameters, expert judgement is needed. But since the parameters values can be easily varied in

the model, the effect of different assumptions on the outcome can be easily investigated and the analyst gets a clear feedback on the effect of different assumptions. Therefore, it should only be seen as a tool assisting but not replacing the analyst. A comparative example was presented showing that it is of utmost importance for the inspectorate to have the ability to detect undeclared processing.
Future work will, inter alia, comprise the enhancement of the cost model, the additional visualization of the game theory results and the operationalization as well as the integration into existing systems at the IAEA.

# 5 Acknowledgments

This paper was prepared as an account of work sponsored by the Government of the Federal Republic of Germany within the Joint Programme on the Technical Development and Further Improvement of IAEA Safeguards between the Federal Republic of Germany and the IAEA.

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# Nomenclature

| | |
|---|---|
| AP | additional protocol |
| APA | acquisition path analysis |
| CSA | comprehensive safeguards agreements |

| | |
|---|---|
| NNWS | non-nuclear weapon State |
| NPT | non-proliferation treaty |
| NWS | nuclear weapon State |
| PC | proliferation cost |
| PT | proliferation time |
| SIR | Safeguards Implementation Report |
| SLC | State-level concept |
| TD | technical difficulty |
| VOA | voluntary offer agreement |

# ESAM III – the latest release of Euratom's seal handling application

**Stefano Ciccarello, François Loddo, Peter Schwalbach**

European Commission – Directorate-General for Energy
Direction E – Nuclear Safeguards
1, rue Henry M. Schnadt
Zone d'activités Cloche D'Or
L-2530 Luxembourg
E-mails: stefano.ciccarello@ec.europa.eu, francois.loddo@ec.europa.eu,
peter.schwalbach@ec.europa.eu

***Abstract:***

*The objective of the ESAM (Euratom Seals Analysis and Management) application is to assist Euratom inspectors in managing seals placed on a daily basis in all nuclear facilities throughout the EU. ESAM records all different events which occur during a seal life cycle: Production, Placement, Detachment and Verification.*

*Since its first launch almost 30 years ago, ESAM is now on its third major release which enables:*

- *Handling all types of seals such as Metallic, Optical, Electronic and Ultra-sonic;*
- *Handling a generic seal life-cycle workflow which can be applied for any kind of seal;*
- *Bringing it on site and use it as an In-Field application so to:*
  - *record all inspection's seal events and confinement modifications*
  - *generate reports and working papers related to inspection*
  - *synchronise with HQ database once inspectors are back in Euratom premises.*
- *Providing menus and actions according to user's role (Inspector, Lab technician and International Agreement Officer).*
- *A full tracking of seals-related events with respect to the seals itself, confinement and inspection.*
- *Providing one structured confinement catalogue with 4 distinct categories.*

*ESAM III is not just one single and consistent database for seals used in controlled nuclear areas, for storage containers, safeguards equipment and shipments of nuclear material. It also enables inspectors to use a "mobile" version on-site therefore removing the need of local tailor-made tools, double entry and related inconsistencies.*

**Keywords**: EURATOM, seal, ESAM, seal life cycle, inspection

## 1. Introduction

The objective of the ESAM (Euratom Seals Analysis and Management) application is to assist Euratom inspectors in managing seals placed on a daily basis in all nuclear facilities throughout the EU.

A first version was launched in the 80's, and with a major release issued every 15 years, ESAM is now on its third version. Initially born to uphold the metallic seals management, it counts now 30 years of history and has evolved alongside the development of the containment techniques: ESAM III is also including optical, electronic and ultrasonic seals management.

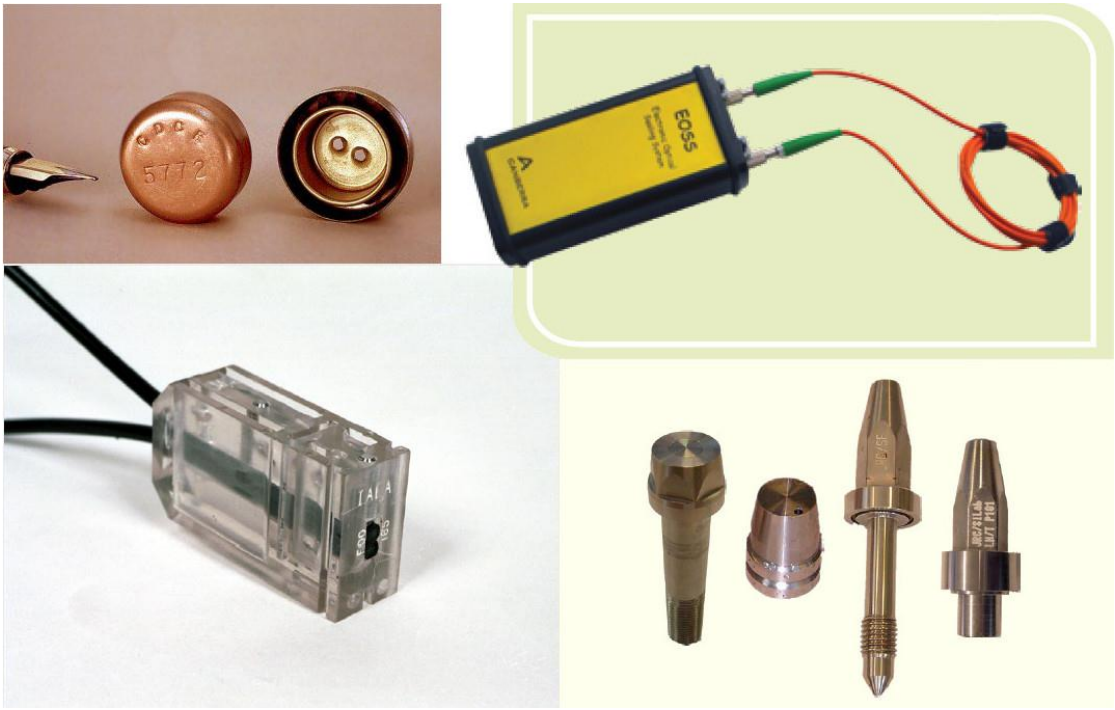



**Figure 1.** Seals handled by ESAM III

Built on the experience acquired by operating the past two releases, ESAM III also interprets the need of an effective use of resources.

## 2. ESAM III: a flexible tool

### 2.1. Upholding different users, tasks and seals' life-cycles

ESAM records all different events which occur during a seal life cycle: Production, Issue, Placement, Detachment and Verification.

Depending on seals type and event, seals handling requires the intervention of different categories of users:

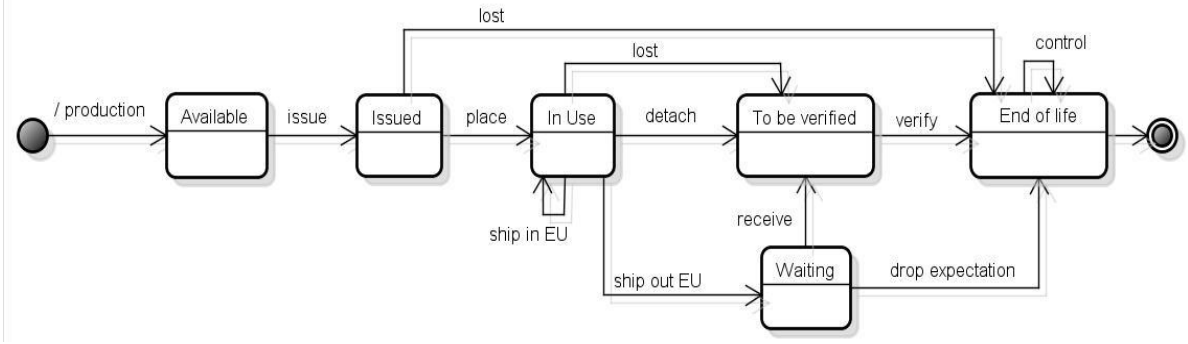
- Production and issue, always performed by Euratom lab technicians
- Placing, always done by inspectors,
- Detachment, might be accomplished by EURATOM or IAEA inspectors, as well as nuclear operators
- Verifications, *in-situ* by Euratom or IAEA inspectors, at headquarters by lab technicians.

In ESAM III different menu actions are made available on the basis of the user's role and seal type. Moreover inexperienced users may use features in wizard instead of expert mode.

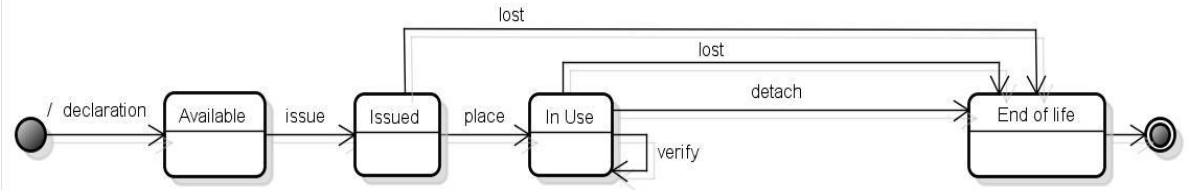
Sensible flexibility is implemented with respect to the seals' life-cycle:

- Single verification and single use for metallic seals,
- Multiple verifications and single use for optical and ultrasonic, and
- Multiple verifications and reusable for the electronic ones.

**Single verification, single use**




**Multiple verifications, Single use, (Optical,**




**Multiple verifications, Multiple use**

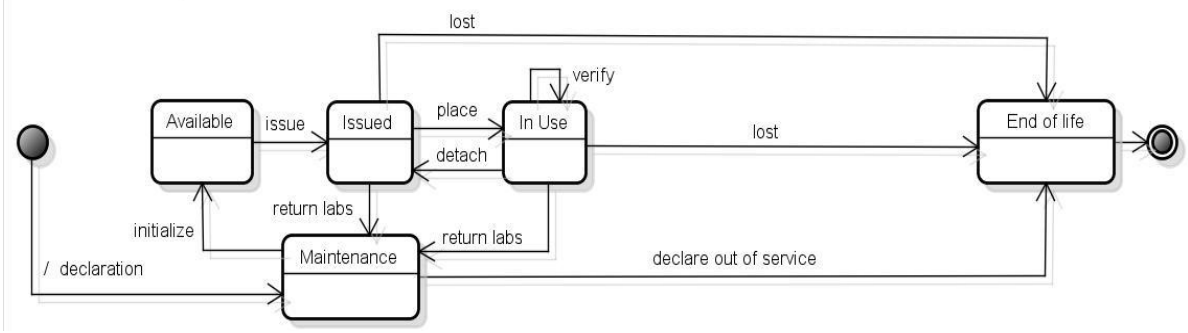



**Figure 2.** Seals' life cycles

Moreover ESAM III provides for events update in a pragmatic mode thereby allowing certain actions to be entered in an "unnatural succession" (e.g. detachment before placing), while, at the same time, keeping anomalies under control.

## 2.2. Confinements

The combination of a great number of different confinements EU-wide with the multilingual aspects of Euratom staff had led to multiple ways of identifying the sealed/sealable "object", resulting in significant difficulties to find and properly track the confinement history in the database.

The latest release of Euratom's seal handling application now provides for a structured confinement catalogue with four distinct categories: areas, equipment, container, sample. Each of them is further defined by a closed list of types (e.g. types for a container could be *drum*, *transcont 20*, *cylinder 48y*, etc). Each couple of category-type allows for a specific name template through a regular expression.

Connected to the confinement category-type couple, the possibility of having a related picture facilitates the recognition of the sealable object concerned.

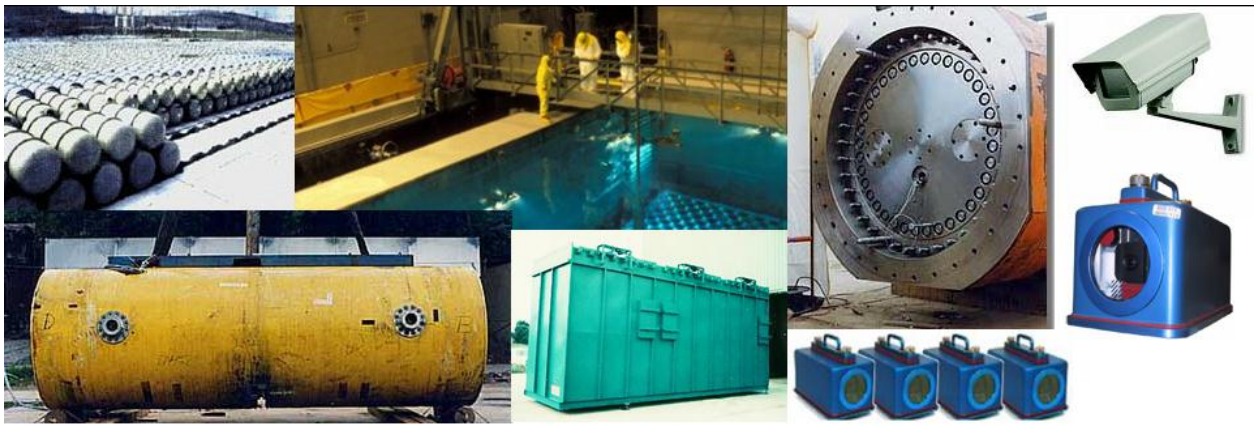


**Figure 3.** Different confinements

### 2.3. An ubiquitous system

ESAM III allows for working there where the seals events take place.

A generic MMS (Mobile Management System) has been built and adopted throughout all Euratom applications. Upon configuration of one or more inspections, users can request to the central system and immediately obtain the production of an *"Inspector kit"*, a single deployable file, bundle of applications and data concerning the inspection(s) context. The kit is deployable on a portable computer or an in-field PC and the inspector may work likewise in the Euratom headquarter premises.

At the end of the inspection(s) an "*Outcome kit",* a file with the updates performed on site, is generated and, once at headquarters, synchronised with the central database. Inconsistencies are highlighted.

## 3. Conclusions

ESAM III is a management system for all kinds of seals used in controlled nuclear areas, for storage containers , safeguards equipment and shipments of nuclear material.

It is one single and consistent database. It has standardized confinements with the support of pictures. A flexible and guided interface together with anomalies follow up contributes to raising the overall performance of the Euratom seals handling system.
This translates into a better tracking. ESAM III provide for historical view of events in relation with all of the three different access points being seals, confinement or inspection.

It enables inspectors to use a "mobile" version on-site therefore removing the need of local tailor-made tools, double entry and related inconsistencies. Already during the first year of operations more than 25% of inspectors' seals events have been introduced through the "Outcome kit". In-field versions of inspectors-made tools have been and are being progressively replaced hence letting inspectors to focus on their core tasks.

# Categorization of the Main Techniques of Neutron Coincidence/Multiplicity Analysis

**Alan Michael Bolind**

Integrated Support Center for Nuclear Nonproliferation and Nuclear Security
Japan Atomic Energy Agency
Funaishikawa Eki Higashi, Tokai-mura, Naka-gun, Ibaraki-ken 319-1118, Japan
E-mail: alan.bolind@jaea.go.jp

***Abstract:***

*Reactor noise analysis and neutron coincidence/multiplicity analysis for safeguards both examine the non-random clustering of neutrons in time, caused by the simultaneous emission of multiple neutrons from each fission event and by the correlation of successive fissions in the same fission chain. Despite these same physical origins, the various kinds of techniques are not often compared with each other on an intuitive level. This paper makes this comparison, primarily by examining whether a given technique relies on the joint probability of detecting neutrons or on the conditional probability. The techniques are compared in other ways, too, such as how they deal with fission chains. Subtleties are also addressed, such as the time origin of the Rossi-α diagram and the taking of moments. Specific instruments are not described in detail; the focus is the understanding of the physics and counting principles.*

**Keywords:** neutron; coincidence; multiplicity; reactor; noise

## 1. Introduction

Neutron coincidence counting is based on the fact that several neutrons are emitted simultaneously from each fission event. The number of neutrons emitted from a single fission event is called the neutron *multiplicity*, ν, which should not be confused with neutron *multiplication*. The number of neutrons varies with each fission event according to a well-known probability histogram, P(ν), for each isotope. (See Figure 1.) (The term *neutron coincidence* will be used in this paper to refer to all orders of analysis, not just first-order analysis.)

The simplest and most ideal kind of measurement is a deterministic one, in which a small amount of a pure but unknown material that produces neutrons only by spontaneous fission is put into the center of a chamber that is surrounded by high-efficiency neutron detectors. The combined hardware of the chamber, the detectors, and other components will be called the “instrument” in the following discussion, with the specific exclusion of the sample chamber; the volume of the sample chamber is not part of the instrument. The amount of material put into the sample chamber is kept small so that the activity will also be small, such that most of the spontaneous fission events do not overlap, even though they occur randomly in time. The small amount of material also prevents the neutrons that are emitted from being reabsorbed in the sample due to neutron capture. As for neutron leakage, the high efficiency of the neutron detectors ensures that almost all of the neutrons from each spontaneous fission event are counted.

A histogram of the neutron counts of the fission events is made and is compared with the histograms in the literature (e.g., Figure 1). The radioactive isotope in the sample must be the isotope with the histogram that most closely matches the measured histogram. Because the literature histograms are normalized to a total area of one to make them probabilities, this comparison is a comparison of the shapes of the histograms, not of their scales. After the isotope has been identified, the next question is how much of the isotope is present in the sample; the sample might contain other, non-radioactive isotopes, such that a measurement of the sample’s mass would not represent the mass of the radioactive, neutron-emitting isotope. The non-normalized, measured histogram represents a certain

number of neutrons counted per unit time, and this absolute scale of the histogram can be correlated to the strength of the source, i.e., the spontaneously fissioning isotope's activity and mass. Thus, both the unknown neutron-emitting isotope and its unknown activity and mass can be determined by measuring the neutron multiplicity distribution coming from the sample.

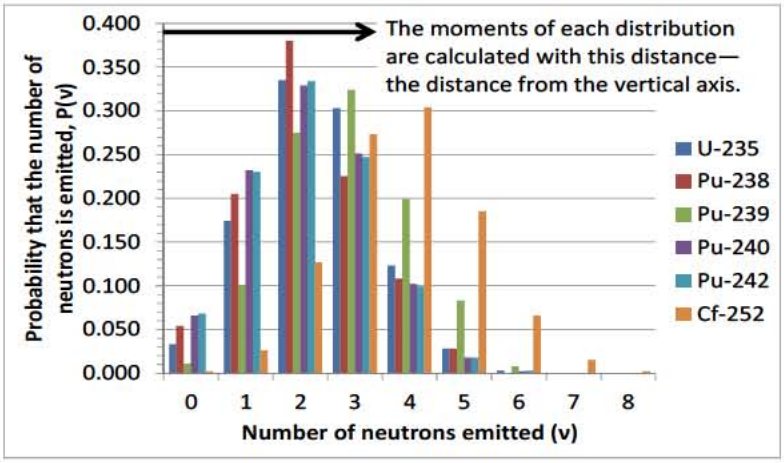



Figure 1: The neutron multiplicity histograms of several actinide isotopes. (The data are from Reference [1].)

In practice, such a simplistic measurement is rarely made, because many more unknowns exist, such as are in the following list:

Unknowns about the neutron sources in the material

- The sample may not be isotopically pure; more than one spontaneously fissioning isotope may be present in the sample. The measured histogram would then be a superposition of those of the various isotopes, weighted according to their activities in the sample.
- Fissile isotopes may be present in the sample, so that induced fissions might occur. Like spontaneous fissions, these induced fissions emit several neutrons at a time, so that, on an event-by-event basis, they cannot be distinguished from the spontaneous fissions except by their multiplicity histograms, as in Figure 1.
- The probability or rate of induced fission depends not only on the amount and spatial distribution of the fissile isotopes. It also depends on the spatial and energy distributions of the neutron flux, which in turn, depend on the sample's geometry, composition, and density. It might also depend on the structure and composition of the instrument—specifically, the reflection of neutrons back into the sample by the instrument.
- Additional neutrons may be emitted by non-fission processes, such as by (α,n) reactions. These neutrons are emitted one at a time, which would ordinarily allow them to be distinguished from the spontaneous fissions, which are emitted several at a time. However, the intensity of the (α,n) neutron source is often high enough that these neutrons overlap with those from the fission events. Moreover, the singly emitted (α,n) neutrons can induce fission in fissile isotopes, which then emit several neutrons simultaneously. With sufficiently high multiplication in a sample, the fissions induced by the (α,n) neutrons can thus even be a more dominant concern than the detection of the (α,n) neutrons themselves.

Unknowns related to the inability to detect all of the neutrons

- Some neutrons may be captured in the sample or in non-sensitive parts of the instrument, such as in the polyethylene moderator surrounding a detector. Although this second possibility can be minimized or fixed by the design of the instrument, the first possibility—capture in the sample—depends on the composition of the sample, which is often unknown. It also depends

on the geometry and density of the sample. Moderating material in the sample (i.e., lighter elements, especially hydrogen) has a major influence because it brings down the energy of the neutrons, making them more likely to be captured.

- Some of the neutrons may leak out of both the sample and the instrument without being detected.

Unknowns regarding the interpretation of the neutrons that are detected

- The activity of the sample may be high enough that the spontaneous fissions, induced fissions, and (α,n) reactions overlap.
- More specifically, the time between when the neutrons are emitted and when they are detected—that is, the physical detection time—is not instantaneous but has a finite duration that depends (1) on the geometry of the sample and the instrument and (2) on the moderation of the neutrons to low energy, at which they can be detected with high probability (high efficiency). In other words, there is a finite flight time between the position of the neutron-emitting event and the detectors, plus there is a finite moderation time that is a function of the kinds of scattering events that the neutrons encounter. (This moderation time also depends on the energy with which the neutrons are emitted, but often this variable can be ignored.) Even for a given sample and instrument, this physical detection time varies with each neutron, so that the detection of neutrons from one event can overlap with the detection of those from another event, even if the events did not actually occur simultaneously. In short, the physical detection time smears the chronology of the neutron-emitting events.
- From the point of view of the instrument, all the neutrons are indistinguishable from each other except insofar as the time at which they are detected. The overlapping of the detection of neutrons from separate events, due to high radioactivity and to the variance of the physical detection time, removes much of this chronological information. The information that remains is not sufficient for a deterministic analysis, such as the comparison of histograms that was mentioned above for the simplest case. Instead, the detected neutrons must be treated with a probabilistic analysis.

Both the production of neutrons and the detection of neutrons are mixtures of random processes and correlated processes. Primary neutrons are produced by spontaneous fission and (α,n) reactions, which are both forms of radioactive decay and are therefore random processes. Induced fission, which produces secondary neutrons, can be a correlated process, because it requires incident neutrons. Therefore, an induced fission event is always somewhat correlated to the spontaneous fission, (α,n) reaction, or other induced fission that produced that incident neutron. (This correlation of induced fission events will be discussed in more detail below.)

The detection of neutrons is also a mixture of randomness and coincidence. Not only does the detection of neutrons reflect the randomness in their creation, but the variation in the physical detection time also contributes further randomness. On the other hand, each fission event releases several neutrons simultaneously, so that these neutrons are perfectly correlated with each other at the instant of their creation. Thus, even if the fission events themselves are random, the neutrons that each event produces are grouped together in time.

Probabilistic analysis deals with these randomizing influences by examining, within a small time period, either when the neutrons detected or how many neutrons are detected. This time period, which is often called a “gate,” is small because it is on the same scale as the physical detection time or the die-away time of fission chains; it can vary approximately from one microsecond to many tens of microseconds. The measurement over this gate is repeated many, many times to produce values for averages and other statistical metrics. The gate width (i.e., the length of the time period) is varied over a range in some types of analyses. In this case, separate measurements of the neutron pulse train (i.e., the chronological detection of the neutrons) can be made with different gate widths, or the same neutron pulse train can be stored electronically and analyzed repeatedly with various gate widths.

There are two main categories of probabilistic analyses: those by joint probability and those by conditional probability. Joint-probability analyses examine how many neutrons are detected within a gate that is begun ("triggered") at random or periodic times. Figure 2 illustrates this process for two different gate widths. The key feature of joint-probability methods is that the starting times of this gate must be random with respect to both the emission of neutrons and their detection. So, the period for counting neutrons often begins in the "blank space" between the detections of neutrons. Conditional-probability analyses, in contrast, examine only the subsequent neutrons that are detected after an initial neutron is detected. Before this first neutron is detected, the equipment (or data-analysis routine) merely waits, doing nothing. If the neutron flux is purely random and without any correlation, then the average number of neutrons counted within a gate of a given width is the same for both types of analyses; but if there is correlation, then the conditional-probability analyses count more neutrons than the joint-probability analyses. These properties can be seen in the definition of conditional probability for two detections, at times $t_1$ and $t_2$:

$$P(t_2|t_1) = \frac{P(t_1 \cap t_2)}{P(t_1)} \qquad \text{Equation 1}$$

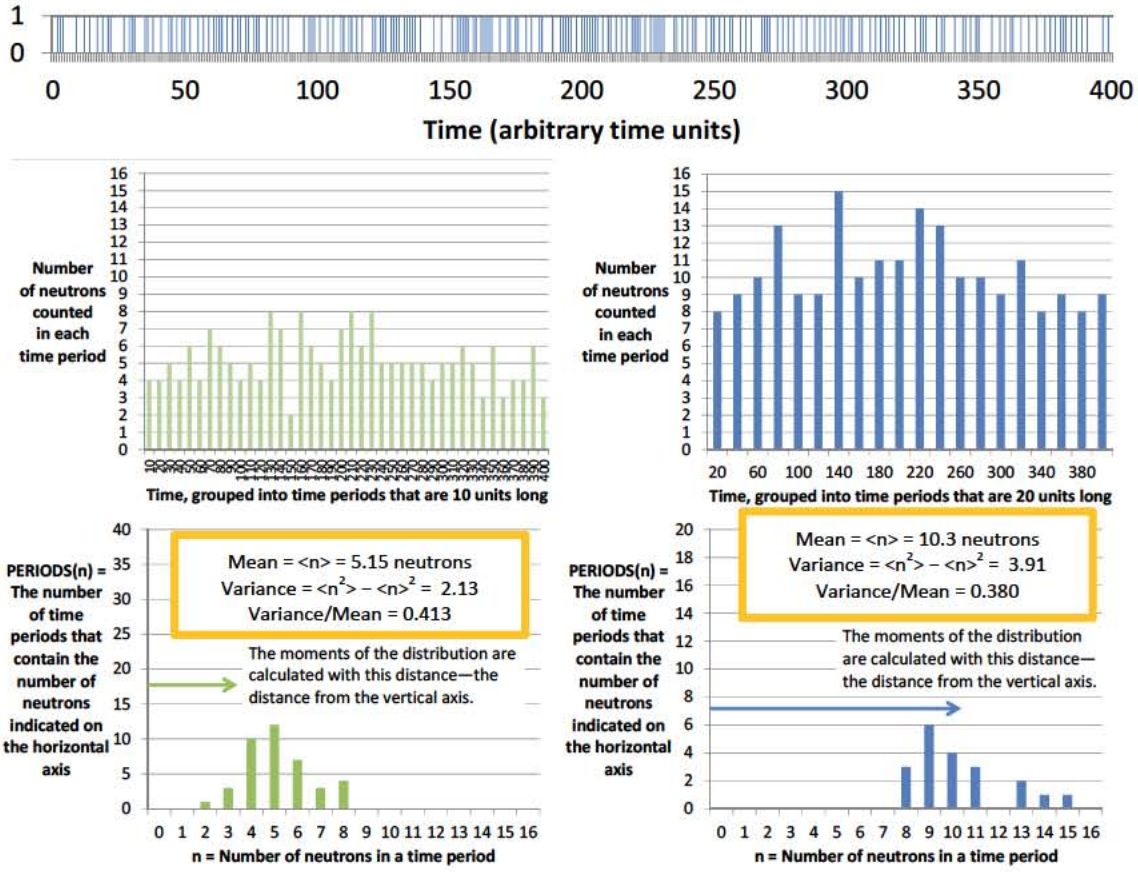



Figure 2: An arbitrary neutron pulse train (top); groupings of these neutrons into time periods of two different lengths (10 time units, middle left; 20 time units, middle right); histograms of those groupings (bottom left and right)[1]

If the detections are purely random and independent, then the following relations hold true:

$$P(t_1 \cap t_2) = P(t_1)P(t_2) \text{ for independent events}$$

$$\text{so, } P(t_2|t_1) = \frac{P(t_1)P(t_2)}{P(t_1)} = P(t_2) \text{ for independent events} \qquad \text{Equation 2}$$

[1] Figure 2 was created not by neutron detections but by the author tapping his finger randomly on his computer keyboard. This is why the variance-to-mean ratio of the 20-time-unit analysis is less than that of the 10-time-unit analysis, when it should be greater (Figure 3).

The crucial importance of this randomness in the triggering of joint-probability analyses demands further contemplation. There are three ways to achieve this randomness.

Firstly, the triggering source (i.e., whatever entity that decides when to start counting) could be external to both the instrument and the sample being measured. For example, the detection of photons from the radioactive decay of a separate cesium source in a neighboring building would be a suitable external, triggering source. Whenever a photon would be detected, then the NDA instrument would begin measuring neutrons and would continue to do so until the end of the gate, at which point it would stop and wait for another trigger from the photon source. Although it is theoretically satisfying, such an experimental setup is obviously excessively complicated in practice.

The second way to achieve randomness is much easier; it is simply to start the measurements at regular time intervals. An example is the starting of the next gate as soon as the previous gate finishes. Since the neutrons are originally created by radioactive decay (i.e., by spontaneous fissions and (α,n) reactions), and since radioactive decay is inherently random with respect to linear time, the starting of the gates at regular, periodic times is indeed random with respect to the neutrons. Because of its simplicity, most joint-probability analyses rely on this periodic triggering.

The third way to achieve random triggering is the most subtle one, but it is also an essential concept for understanding the shift-register method (Section 3). In this third way, the detection of a neutron from one fission chain serves as a random trigger for the detection of neutrons from other, separate fission chains. The independence of the fission chains makes this kind of triggering possible, and this independence is, in turn, caused by the randomness of the primary neutron sources. Specifically, the events that create the primary neutrons are not just random with respect to time, but they are also random *with respect to each other*. Thus, this third way of achieving randomness is almost identical to the first way, except that the random photon source in the neighboring building has been replaced by a random neutron source in the same sample. A complication, though, is that, although primary-neutron creation events are instantaneous, physical detection times and fission chains both have finite durations and so can overlap in time. Therefore, the only way to ensure that the triggering neutron and the measured neutrons are indeed from separate fission events and chains is to *wait* for a certain period of time after the triggering neutron, until the neutron population from the fission event and chain of which it was a member has most assuredly died away (terminated). Only then can the gate begin, with the confidence that the detected neutrons will have no correlation whatsoever with the neutron that triggered the gate. This third way of achieving randomness thus opens up the possibility of simultaneously performing both a joint-probability analysis and a conditional-probability analysis; the triggering neutron begins both a conditional-probability analysis in the short term and a delay that starts a joint-probability analysis in the long term. The shift-register method (described below), although it is fundamentally a conditional-probability method, partially incorporates this idea.

In the conditional-probability method, the gate is triggered by the detection of a neutron. In theory, the gate could be triggered by the fission itself, but in practice, it is usually impossible to detect the fission, since in most cases, doing so would require the detection of the prompt gamma rays or some other, super-fast particles (like anti-neutrinos). (The exception is discussed in the next paragraph.) It must be reiterated that the chronology of the fission events is smeared by the neutrons' physical detection time and the overlapping of events. Therefore, if the reader sees a graph in the literature that refers to the dying away of a neutron population with the time *after the fission event*, the reader must realize that such a graph *cannot* be determined directly by experiment, since the exact time of the fission event is always unknown. Instead, the conditional-probability method produces a graph of the probability that a neutron will be detected after the detection of the first neutron (Figure 4, for example). It is this first detection that defines the zero time ($t = 0$), thereby implying that the fission event that produced the neutron must have occurred at some unknown, negative time. This conditional-probability curve—or the non-normalized rate curve that corresponds to it—is known in the literature as the Rossi-α curve, named after Bruno Rossi, who suggested this kind of analysis.[2]

The exception mentioned above, in which a fission event can be directly detected, is when a fission chamber both is used as the neutron detector and is inherently part of the system being measured. A fission chamber located in the middle of a nuclear reactor satisfies this requirement. Any fission event in the fission chamber is detected by its fission fragments, while its neutrons proceed to perpetuate the fission chains in the system. It appears that perhaps de Hoffmann had this concept in mind when he implicitly defined his efficiency variable, ε, as being the efficiency for detecting fissions [3], although he, Feynman, and Serber actually used $BF_3$ detectors in the experiments for their famous 1956

publication.[4] Williams made this definition explicit in his derivation (page 38 of Reference [5]). Mihalczo took this idea one step further and made a fission chamber out of $^{252}$Cf; the spontaneous fissions of the $^{252}$Cf were recorded in the chamber while their neutrons proceeded to start fission chains in the nearby nuclear system.[6, 7] Another, separate neutron detector then detected the neutrons produced by the nuclear system. Obviously, in Mihalczo's experiment, the starting time for the conditional-probability analysis really was the spontaneous fission event in the $^{252}$Cf. In fact, his experiment was much like a differential-die-away (DDA) experiment, in which each spontaneous fission of $^{252}$Cf served as a tiny neutron pulse to actively interrogate the fuel assembly. In common practice, though, and especially for NDA of fuel assemblies, the neutron detectors are outside the sample or system and are often not fission chambers. Therefore, the usual situation is that the time of the fission event is unknown, and the conditional-probability analysis begins with the detection of a neutron.

As was mentioned before, spontaneous fissions and (α,n) reactions do occur purely randomly, but induced fissions do not, because the neutrons produced by one fission event are the catalysts for the subsequent fission events in the fission chain. The average time between fission events, and the average length of the fission chains in a subcritical system are therefore both important characteristics of system with neutron multiplication. The effect of induced fission has traditionally been handled in either one of two ways.

(1) If the system is nearly critical and therefore has high multiplication (as in zero-power reactor noise analysis), the fission chains are long because they contain many fission events and (for thermal-neutron reactors) because the moderation time of the neutrons between fission events also lengthens the fission chains. In this case, the non-random spacing among the fission events of a fission chain contributes additional correlations to the Rossi-α curve, even to the point of dominating the correlation, over the correlation due to the neutron multiplicity of each fission event. Thus, the induced fission lengthens the exponential portion of the Rossi-α curve and is simply measured directly.

(2) On the other hand, if the system is very subcritical and therefore has low multiplication, the fission chains are short and contain only few fission events. This case has two subcases, according to the reason why the system is subcritical.

(2a) If the system is subcritical because there is little moderation of neutrons, then the fast neutron spectrum also implies short physical detection times. In this subcase, the time between the spontaneous fission or (α,n) reaction that began the fission chain and the time of the last induced fission event of the fission chain is so short that all of the neutrons of the entire fission chain practically arrive at the detectors simultaneously, within the variation of the physical detection time. This fact led Böhnel to call these chains "superfissions," since the chains are indistinguishable from single spontaneous fission events with large multiplicities.[8-10] This subcase is representative of measurements of cans of purified nuclear material (like MOX powder).

(2b) The other subcase is when moderation is present but the system is still subcritical because there is a low density of fissile material or much neutron capture or both. The number and length of fission chains are small because of the lack of fissile material, the time between the fission events in a chain is long because of the moderation, and the chance of detecting all the fission events of a chain is small because of the capture. Thus in this subcase, the correlation among the induced fission events can be lost, so that they appear to be random. NDA measurements on some spent fuel assemblies that are immersed in cooling water might fall into this subcategory.

It is important to clarify the way in which the sample or system (e.g., a fuel assembly) must be subcritical in order for the correlations to be able to be detected. In general, sub-criticality is an essential requirement for coincidence/multiplicity analysis, because if the fission chains never die out, then it becomes impossible to distinguish one chain from another. Succinctly put, if every fission event is correlated, then no fission event is correlated, in a practical sense. In such a system, only the correlation among the simultaneously emitted neutrons from each fission event would, in theory, remain distinct and able to be detected. In fact, though, it is possible to fully analyze a system that is critical on delayed neutrons but is subcritical on prompt neutrons (e.g., a reactor). In such a system, the relatively slow emission of the delayed neutrons causes them to appear more or less like a randomly generated, primary neutron source to the coincidence measurement. Also, the prompt fission chains do, in fact, die away. (If it is necessary, the effects of the delayed neutrons can somewhat be taken into account through more complicated analyses; see Pacilio, for example.[2])

Thus, coincidence/multiplicity analysis can be applied to critical nuclear reactors at zero or very low power levels at which the neutron flux does not overwhelm the detectors. Although this point is not of practical importance for the analysis of completely subcritical, single fuel assemblies, it is nevertheless of conceptual importance when adapting the methods from the reactor-noise-analysis literature to such an NDA application.

## 2. Coincidence analysis by the joint probability of detecting multiple neutrons in a random or periodically triggered time interval

The joint-probability method and the conditional-probability method will now be explained and related to each other. In the joint-probability method, many measurements are made of the number of neutrons that are detected within a gate. This number will be denoted by the letter *n*, and it varies, of course, from measurement to measurement. A histogram can be created from these measurements, showing the number of time periods (gates) that contain each number of neutrons. This number of time periods will be denoted symbolically by the capitalized word PERIODS. Figure 2 illustrates these concepts with an example, in which the same, arbitrary pulse train (top graph) is analyzed according to two different gate widths. Note that these gates are periodically triggered, with the next gate beginning as the previous gate ends. The net effect is to take the continuous time axis of the pulse train (top graph) and make its resolution poorer, by chopping it into consecutive but discrete time intervals.

The moments of the histograms (bottom of Figure 2) can be calculated as follows:

$$j^{\text{th}}\text{moment about the origin} = E[n^j] \text{ or } \langle n^j \rangle \text{ or } \overline{n^j} = \frac{\sum_{n=1}^{\infty} n^j \text{PERIODS}(n)}{\sum_{n=1}^{\infty} \text{PERIODS}(n)}$$

Mean = 1st moment about the origin $= \langle n \rangle$ Equation 3

Variance = 2nd moment about the mean $= \langle (n - \langle n \rangle)^2 \rangle = \langle n^2 \rangle - \langle n \rangle^2$

The first moment about the origin is the average number of neutrons that are expected to be counted during a time interval (gate) of a particular length. Obviously, if the time interval is lengthened (the gate is widened), then more neutrons are expected to be counted. Also, dividing the number of counted neutrons by the duration of the counting is exactly the definition of the average neutron count rate. These concepts are expressed by the following proportionality:

$$\langle n \rangle \propto F(t_2 - t_1) \quad \text{Equation 4}$$

where *F* is the production rate of neutrons (i.e., the rate of fission plus (α,n) reactions), $t_1$ is the starting time of the gate, and $t_2$ is the ending time. Note that the constant of proportionality would have to account for the average number of neutrons per fission or (α,n) reaction (i.e., $\bar{\nu}$) and for the probability that a neutron survives the journey to the detector and is detected (characterized by ε). Therefore, the product of $\bar{\nu}$ and ε is often used for this constant, but ambiguity exists in the literature.

If the arrival of the neutrons at the detector would be completely random, then the probability of detecting a certain number neutrons (say, *k* neutrons) within a specified time interval would be described by the Poisson distribution:

$$\text{P}(n = k) = \frac{\lambda^k e^{-\lambda}}{k!} \quad \text{Equation 5}$$

Here, λ is a parameter that is equal to the mean, <*n*>; this fact can be verified by calculating the mean of this distribution (Equation 5), which will yield λ. Another interesting fact about the Poisson distribution is that the variance is also equal to λ; that is, the variance is equal to the mean. Thus, one way to verify that a process is completely random is to measure the mean and the variance of the output; if they are equal, then the process is random. Conversely, the extent to which the variance is *not* equal to the mean is a measure of the non-randomness of the process—that is, the coincidence among the outputs of the process.

These realizations led De Hoffmann, Feynman, and Serber to determine the ratio of the variance to the mean, of the neutrons counted in a time interval, as a function of physical, neutronic characteristics of the system that is producing the neutrons.[3, 4] Their equation has been derived by several authors and is as follows: [2, 3, 5, 11, 12]

$$\frac{\text{Variance}}{\text{Mean}} = \frac{\langle n^2\rangle - \langle n\rangle^2}{\langle n\rangle}$$

$$= 1 + \frac{\varepsilon\langle \nu(\nu-1)\rangle}{\alpha^2 \tau_f^2}\left\{1 - \frac{1 - e^{-\alpha\Delta t}}{\alpha\Delta t}\right\} \qquad \text{Equation 6}$$

$$= 1 + Y \quad (\text{explained below}) \qquad \text{Equation 7}$$

The time interval is Δ*t*, the average number of pairs of neutrons emitted per fission event is $\langle \nu(\nu-1)\rangle/2$, the average time between fission events is $\tau_f$, and α is the die-away constant that describes how long it takes for the neutrons created by a fission event to disappear, by leakage or absorption. The characteristic lifetime of neutrons in the system is therefore 1/α. The factor ε represents a detection efficiency, but its meaning is ambiguous in the literature and will not be clarified here.

There are seven important features of Equation 6 to highlight:

1. The first term on the right-hand side is one, and it represents the Poisson randomness. The second term represents the extent to which the neutron pulse train is non-random and deviates from the Poisson distribution. This second term is often denoted by the capital letter *Y*, as in Equation 7.
2. The emission of several neutrons from each fission event (i.e., the neutron multiplicity) is represented by the factor $\langle \nu(\nu-1)\rangle$.
3. The mean, the variance, and the variance-to-mean ratio depend upon the length of the time interval (width of the gate; Δ*t*). See Figures 2 and 3.
4. As the gate becomes wider, the variance-to-mean ratio does *not* go to one. Only the term in parentheses goes to one, thus preserving the second term yet also saturating the variance-to-mean ratio.
5. As the gate becomes smaller, the second term goes to zero, reflecting the fact that the physical detection time (represented through α) and the time between fissions (represented by $\tau_f$) smear the chronology of the neutrons and make it less likely to detect correlated neutrons within a gate that is too narrow.
6. As the time between fissions, $\tau_f$, becomes longer, the second term becomes smaller and the variance-to-mean ratio becomes closer to one. This feature is partly a reflection of the physics mentioned earlier, that a long and varied time between induced fission events removes the correlation between those events. It is primarily caused, though, by the original purpose for which Equation 6 was derived—namely, for the detection of neutrons from *fission chains*, rather than for the detection of neutrons from individual fission events. The early authors were concerned about measuring the reactivity of nuclear reactors that were close to criticality.[2-4] In this view, the primary purpose of a fission event is to perpetuate the fission chain and thereby make it more likely that a neutron from that particular fission chain will be detected. This reason is why $\tau_f$ is multiplied by α in the denominator, because a short time between fissions counteracts a fast die-away time (a large value of α).

It should also be noted that delayed neutrons have been neglected in this equation, and only prompt neutrons have been considered.

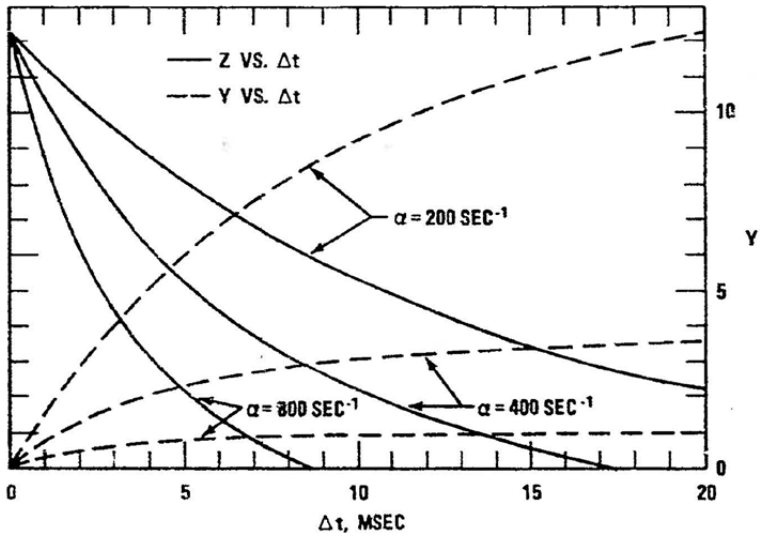



Figure 3: a comparison between Z (from the mean number of pairs in a time interval) and Y (from the ratio of the variance to the mean), as a function of gate width and for various values of α (the die-away time). The graph is for a specific set of experimental parameters. Copied from Reference [2].

The traditional way to use Equation 6 is to determine the variance-to-mean ratio for several values of $\Delta t$, which requires either several measurement runs or several re-analyses of one measurement run. The resulting values are plotted as a function of $\Delta t$ and are fit with Equation 6 by least-squares fitting, to determine unknown quantities. In reactor noise analysis, the usual goal is to find the value of α for the reactor, because it can be related to its reactivity. The value of $\langle \nu(\nu-1) \rangle$ is estimated or known. In contrast, the usual goal in safeguards NDA is to find the value of $\langle \nu(\nu-1) \rangle$, because it is characteristic of the unknown isotopes in the sample (Figure 1). The die-away time (α) and the efficiency (ε) are controlled by the designs of the instrument and the sample and are determined by calibration.

The variance-to-mean ratio is directly related to the rate of counting two neutrons within $\Delta t$ and can be derived algebraically from Equation 6: [5, 12]

$$\text{Mean number of pairs} = \left\langle \binom{n}{2} \right\rangle = \left\langle \frac{n(n-1)}{2} \right\rangle = \left(\frac{1}{2}\right)\{\langle n^2 \rangle - \langle n \rangle\}$$

$$= \frac{[F\varepsilon(t_2 - t_1)]^2}{2} + F\varepsilon(t_2 - t_1)\left(\frac{\varepsilon}{\alpha^2 \tau_f^2}\right)\left(\frac{\langle \nu(\nu-1) \rangle}{2}\right)\left\{1 - \frac{1 - e^{-\alpha(t_2 - t_1)}}{\alpha(t_2 - t_1)}\right\} \qquad \text{Equation 8}$$

$$= \frac{[F\varepsilon(t_2 - t_1)]^2}{2}(1 + Z) \quad \text{(explained below)} \qquad \text{Equation 9}$$

The first term on the right-hand side of Equation 8 is the mean number of pairs to be expected from a Poisson distribution, as manipulations of Equations 4 and 5 can show:

$$\text{Variance} = \langle \mathrm{n}^2 \rangle - \langle \mathrm{n} \rangle^2$$

$$\text{Variance} = \text{Mean} = \lambda \text{ (for a Poisson distribution)}$$

$$\lambda^2 + \lambda = \langle \mathrm{n}^2 \rangle$$

$$\text{Mean Number of Pairs (Poisson Distribution)} = \frac{\lambda^2}{2} = \frac{[F\varepsilon(t_2 - t_1)]^2}{2} \qquad \text{Equation 10}$$

Since the Poisson distribution represents randomness, this first term in Equation 8 is often called the "uncorrelated pairs." The second term is therefore correspondingly called the "correlated pairs." Note

that it is very similar to the second term on the right-hand side of Equation 6, with the time interval (Δt) being written out explicitly.

Furuhashi and Izumi chose to perform joint-probability analysis from this perspective. [13, 14] They subtracted off the first term of Equation 8 and then divided it by Equation 10, to form the ratio of the correlated pairs to the uncorrelated pairs in a joint-probability analysis. Following Feynman's choice of the letter *Y*, they denoted this ratio by the capital letter *Z*: [2]

$$\frac{\text{Average number of correlated pairs}}{\text{Average number of uncorrelated pairs}} = \frac{\left\langle \binom{n}{2} \right\rangle - \frac{\langle n \rangle^2}{2}}{\frac{\langle n \rangle^2}{2}} = \mathrm{Z}$$

$$\mathrm{Z} = \left(\frac{\varepsilon}{\alpha^2 \tau_f^2}\right)\left[\frac{\langle \nu(\nu-1) \rangle}{F\varepsilon\Delta t}\right]\left\{1 - \frac{1 - e^{-\alpha\Delta t}}{\alpha\Delta t}\right\} \qquad \text{Equation 11}$$

(See also Equation 9.) As the length of the time interval goes to infinity, this ratio goes to zero, as can be seen mathematically by taking the limit. Physically, this fact occurs because, as the time interval increases, pairings are increasingly made among neutrons from random fission events and (α,n) reactions, rather than among the neutrons from just one fission event or short fission chain. Since pairings are a combinatorial quantity, the random pairings *among these unconnected fission events* rapidly swamp the correlated pairings within the set of neutrons of each of these fission events. The denominator of Equation 11 thus goes to infinity faster than the numerator, causing the ratio to go to zero.

(Note that this explanation is different from Pacilio's explanation on page 41 of his text, in which he says that the number of correlated pairs saturates within the interval as it is lengthened.[2] On the contrary, each extra fission event that is included does contribute more correlated pairs, since each fission event produces several neutrons simultaneously. The key is that these extra events contribute *even more uncorrelated pairings* than correlated ones, because the events themselves are uncorrelated with each other. Thus, the explanation here, rather than Pacilio's, is the correct one.)

One fact about Equation 11 that is essential to remember is that it is for *joint probability*, which means that these correlated and uncorrelated pairs are the pairs within *randomly or periodically triggered* time intervals. It is possible or even likely that any particular such time interval will contain zero neutrons. In contrast, conditional-probability analysis specifically excludes time periods that contain zero neutrons. The correlated and uncorrelated pairs in conditional-probability analysis are therefore different from those of joint-probability analysis, as presented in Equations 8 through 11.

## 3. Coincidence analysis by the conditional probability of detecting multiple neutrons as a function of the time after detecting one neutron

The fundamental expression of conditional probability is the Rossi-α curve (Figure 4). In its normalized form, it is the probability density function for a neutron to be detected after a first neutron has been detected. In its non-normalized form, it is the rate at which neutrons are detected after a first neutron has been detected, with units of counts per second. Figure 4 is not normalized, for example. The reader must be warned that almost all the literature uses the non-normalized form while simultaneously calling it a probability, which is obviously false.

The Rossi-α curve can be found either by an independent derivation or by manipulating the equation for the mean number of pairs in a randomly or periodically triggered time interval, Equation 6. This latter option has the advantage that the Rossi-α curve and the variance-to-mean ratio are thereby clearly proven to be related.

Taking the derivatives of Equation 6 with respect to the first and second detection times gives a kind of "joint" rate. By analogy with the definition of conditional probability, the average rate at which neutrons are detected (at time $t_2$) after a first neutron (at time $t_1$) is detected is as follows:

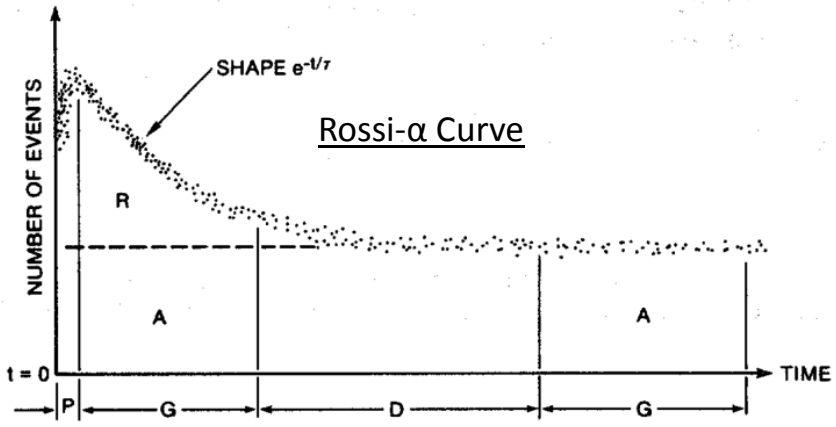



Figure 4: A sketch of a one-dimensional Rossi-α curve, which shows the rate at which second neutrons are detected as a function of the time after first neutrons are detected (which is the origin, $t = 0$). Note that the dots in the curve have no real meaning. Copied from Reference [1]

$$Rate(t_2|t_1) = \frac{\left|\frac{\partial}{\partial t_1 \partial t_2}\left\{\left\langle\binom{n}{2}\right\rangle\right\}\right|}{F\varepsilon} = F\varepsilon + \left(\frac{\varepsilon}{\alpha\tau_f^2}\right)\left(\frac{\langle\nu(\nu-1)\rangle}{2}\right)e^{-\alpha(t_2-t_1)} \qquad \text{Equation 12}$$

by analogy with conditional probability, $P(t_2|t_1) = \frac{P(t_1 \cap t_2)}{P(t_1)}$.

With $t_1$ set at zero and $t_2$ being variable, Equation 12 is the one-dimensional Rossi-α curve for prompt neutrons. [2, 5, 15] The rates are measured experimentally; and ε and $\tau_f$ are assumed or separately measured. A least-squares fit of the *Rate*($t_2|t_1$) data yields α; then Equation 12 is solved for $\langle\nu(\nu-1)\rangle$, which is compared with the value from the literature to determine the isotopes in the sample.

There are two main ways to create a Rossi-α curve from measurements. (See Pacilio's report, page 13 [2], and Seifritz and Stegemann's paper, page 139 [7].) In both ways, the instrument waits until a neutron is detected and only then begins recording data. After the preset measurement time has elapsed, the instrument stops recording data associated with that first neutron and waits instead for another "first" neutron detection to trigger it again. The first way to make measurements is to not overlap measurement intervals. Even though several neutrons may be detected during the measurement period that was triggered by a first neutron, those several neutrons do not themselves start their own, separate measurement periods. Thus, this first way is wasteful of measurement time; because there is no physical distinction between one neutron and another, all neutrons can be considered as "first" neutrons. This fact leads to the second way to make measurements, which is indeed to start separate and overlapping measurement intervals after each and every neutron detection. This second way was actually the original way, used by Orndoff.[2, 7, 16] The two ways actually lead to different results. Babala [17, 18] explained it with reference to the Kolmogorov general theory of chain processes.

This difference between these two measurement methods leads to an important consideration of exactly how a Rossi-α curve comes to have its shape. The question is, "If every neutron can be a 'first' neutron—including the last neutrons to be detected from a fission event—then why is there a hump at the start of the curve? Why is the curve not flat, or why does it not have some random shape?" This question is related to the earlier warning about interpreting graphs that use the time of the fission event as the time origin ($t = 0$) of the graph.

The answer is most easily explained visually with a graph (Figure 5) of the creation of a Rossi-α curve by the second method mentioned above, that of starting a measurement interval with each and every detected neutron. As each neutron is detected, a type of gate is opened, to record the detections of all of the following neutrons within a specified time period. Because a separate gate is opened for each detected neutron, the subsequent neutrons are often being recorded in several gates simultaneously. This type of gate is not the same as the gates in the shift-register method of counting (described below), because those gates do not keep track of when the subsequent neutrons are detected,

whereas this gate does. These time measurements of this gate are intentionally limited in their resolution, though, or else the resolution is worsened during the data analysis. In the example of Figure 5, the neutrons are put into time slots that are each 1 μs wide. The reason is simply that time is a continuous variable whereas neutrons are discrete entities; the discretization of time into time slots is therefore necessary in order to build up a histogram. This process of building up the histogram is illustrated in the middle and bottom-left parts of the figure. The separate gates associated with each detection are shown by stacking the time sequences.

It can be seen in Figure 5 that two kinds of time intervals are involved. The first kind is the differences in the physical detection times of the several neutrons emitted from each fission event. Although the physical detection time has some variation, the variation is nevertheless limited. The second kind is the time between fission events. This time interval is completely random for spontaneous fission events and (α,n) reactions but has some correlation for induced fission events, as mentioned before.

Consider first the presence of only the random events, that is, the spontaneous fissions and (α,n) reactions. During the recording of the data, the relatively close and consistent spacing of the physical detection time causes a buildup in the neutrons counted in the time slots closest to the origin. The random spacing between fission events, on the other hand, evens out the number of neutrons recorded at later times, farther from the origin. It is particularly important to recognize that this evening out occurs even though *clusters* of neutrons are indeed being recorded at these later times, because of the multiple neutrons released from each fission event. The fact that the neutrons are clustered has an effect only at the short time frames, when the several neutrons within each cluster cause a consistent and rapid succession of counts. Thus, the clustering of neutrons from each fission event implies correlation within each cluster but not from cluster to cluster. These principles remain true even when the fission events overlap with each other (which has not been done in Figure 5 for the sake of pedagogical clarity).

When induced fissions are included, the time between fission events has some correlation. This correlation appears in the Rossi-α diagram in the same way as the correlation among neutrons from the same fission event, except that the time scale over which the correlation is present is longer. For example, the exponential part of the histogram in the bottom right portion of Figure 5 would extend to later times when induced fission would be present. Indeed, what is happening is that there is a superposition of two die-away processes: the shorter-lived die-away caused by the variation in the physical detection time, and the longer-lived die-away caused by the subcritical decay of fission chains. If both die-away processes are exponential (as they are usually taken to be), then the superposition must be of the following form:

$$Rate(t_2|t_1 = 0) = F\varepsilon + C_1 \exp\left(-\alpha_{physical\ detection\ time} t_2\right) + C_2 \exp\left(-\alpha_{fission\ chain} t_2\right) \qquad \text{Equation 13}$$

Here, $C_1$ and $C_2$ are constants. (Incidentally, the Differential Die-Away Self Interrogation (DDSI) method [19-21] attempts to distinguish between these two exponential terms.) The evening-out effect that was discussed in the previous paragraph still occurs when induced fission is present, but it can now be considered as being produced by the randomness among separate fission chains. In other words, the random spontaneous fission and (α,n) events, which produce the evening-out effect, now have fission chains attached to themselves.

The shift register is actually a piece of electronic hardware, used for counting neutrons.[1, 2, 7-9, 22, 23] The way that it counts the neutrons, though, is somewhat unique; and so the name for the hardware is often applied to the associated method of counting. The use of shift registers and the shift-register method of analysis is widespread in neutron coincidence and multiplicity analyses in the field of nuclear safeguards.

It is essential to recognize that the shift-register method is a conditional-probability method. Data—namely, the blank times between neutrons—are not recorded until one neutron is detected, thereby triggering the apparatus to record subsequent detections for a preset measurement period. This applies to the so-called “R+A” gate (Region 1 in Figure 6). The so-called “A” gate (Region 2 in Figure 6) is not conditional, since it is randomly triggered as explained in Section 1. Nevertheless, the coincidence information comes from the R+A gate and not the A gate, so the overall method is conditional.

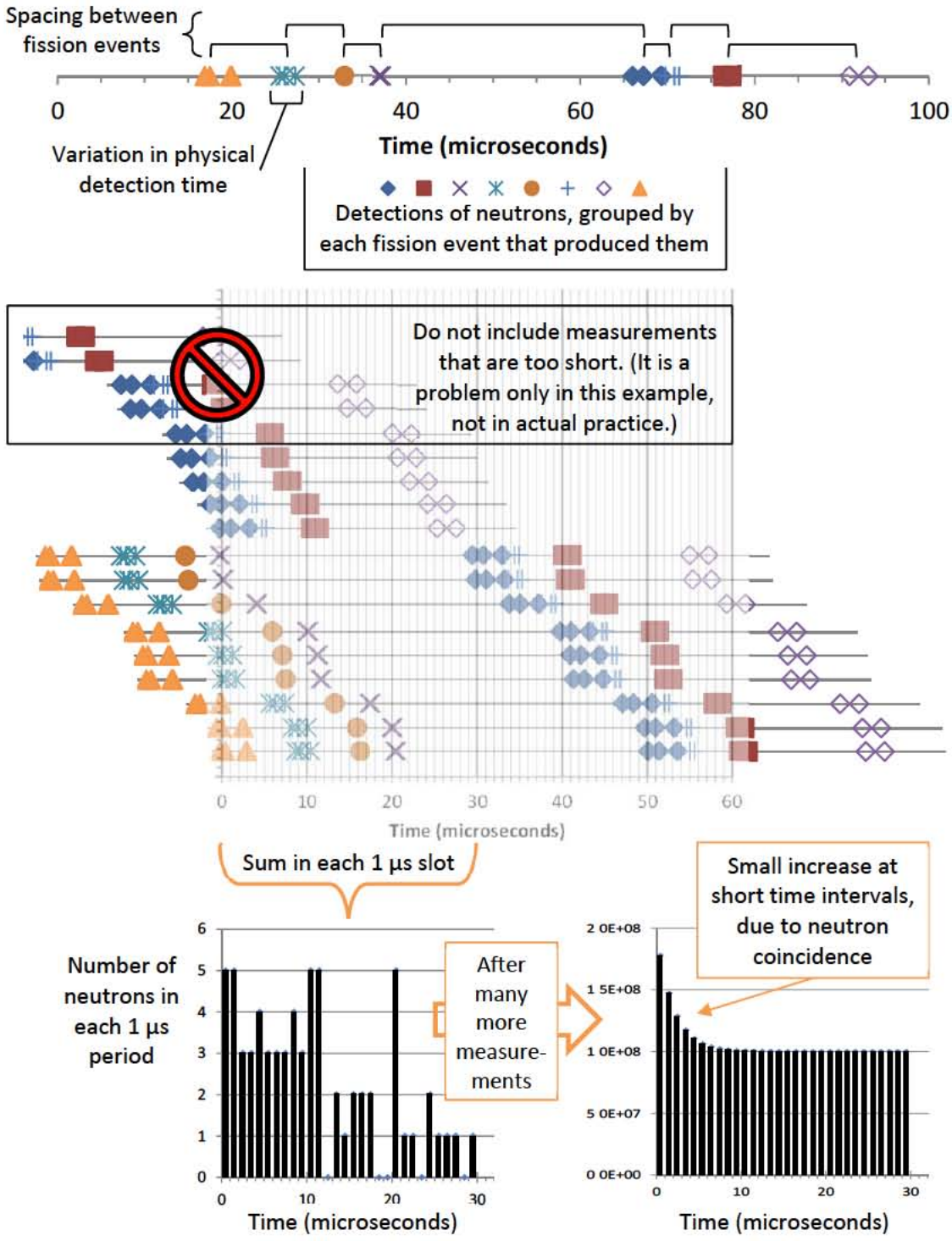



Figure 5: An example of creating a Rossi-α diagram from a time-sequence of neutron detections. Top: the time sequence. Middle: recording the data at each neutron detection, beginning with the first detection (bottom) and going to the last neutron detection (top). Bottom Left: summing the number of neutrons that fall into each 1 µs slot, from 0 µs to 30 µs. Bottom Right: a speculation about the final Rossi-α curve after measuring many more time-sequences (note the change in scale).

The shift-register method of analysis essentially integrates Equation 12 over two separate regions of time: one for early times at which the exponential factor is still significant (Region 1), and one for later times at which it is practically zero (Region 2). (See Figure 6.) The difference between the regions thus corresponds to the second term and therefore to the multiplicity distribution. (See Equation 14.) With each triggering neutron (the "$t_1$" neutron), the number of neutrons within each region is tallied. Then for

each region, a histogram can be made of the number of times it contains a certain number of neutrons, in an almost identical fashion as in Figure 2. (See Figure 7.) In other words, the gate width $\Delta t$ is set to be long enough to span Region 1, but unlike in Figure 2, the time periods begin at each time that a neutron is detected, so that several time periods typically overlap at any given instant. For this reason, the PERIODS variable is given the subscript "S.R." (standing for "shift register") to distinguish it. Moments can be taken of these histograms (Figure 7). Note that it appears that more neutrons are counted in this figure than in Figure 2; the reason is that the time periods overlap, so many neutrons are counted several times each. The shift-register method is thus a hybrid between the "joint-probability" method of the variance-to-mean ratio and the "conditional-probability" method of the Rossi-α method.

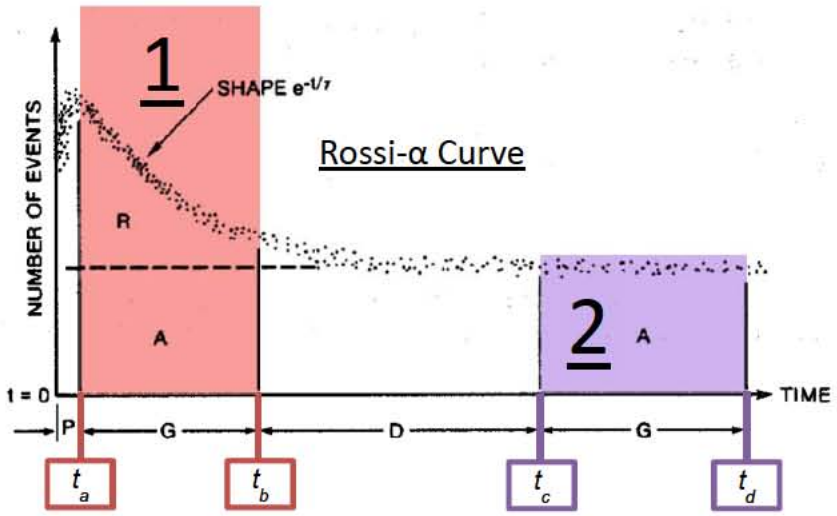



Figure 6: A modification of Figure 4, showing how the shift-register method is an integration of two regions of the Rossi-α curve

$$Region\ 1 - Region\ 2 = \int_{t=t_a}^{t=t_b} Rate(t_2|t_1)dt - \int_{t=t_c}^{t=t_d} Rate(t_2|t_1)dt$$
$$\cong \left(\frac{\varepsilon}{\alpha^2\tau_f^2}\right)\left(\frac{\langle \nu(\nu-1)\rangle}{2}\right)\left(e^{-\alpha t_a} - e^{-\alpha t_b}\right) \qquad \text{Equation 14}$$

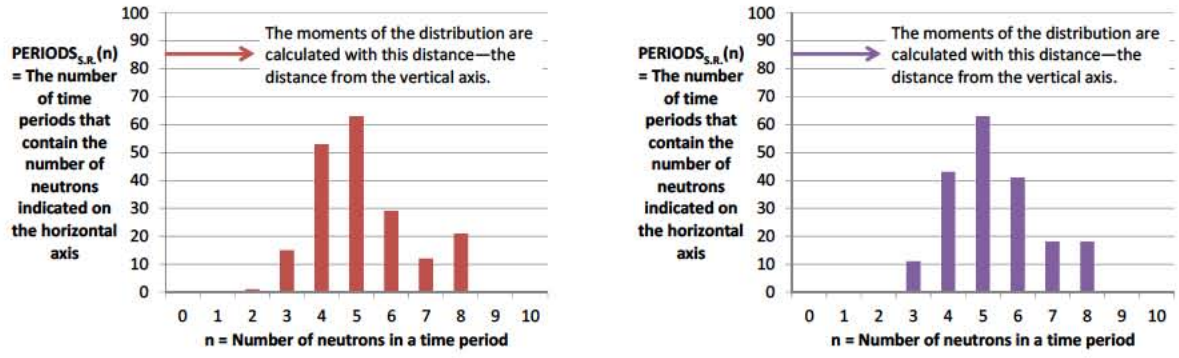



Figure 7: Histograms of the from a shift-register analysis of the neutron pulse train of Figure 2. The gate width is 10 time units long. Left: Region 1 (R+A gate). Right: Region 2 (A gate)

The analysis by Dierckx & Hage [24] considered the differences in the moments of the two histograms for Regions 1 and 2 (e.g., Figure 7). The difference in the first moments (the means) represents the analysis of Equation 14. Dierckx & Hage said that the differences in the higher moments also represent the differences between correlated and uncorrelated multiplets. However, the histogram for Region 2, which Dierckx & Hage said is the "background" because it comes from random triggering,

actually contains some correlation. If Region 2 is truly randomly triggered, then by the analysis of Feynman, De Hoffman, and Serber (Equations 3 through Equation 7), it must contain correlation, otherwise their analysis would not work. The reason that the first-order analysis (first moments) works is that the mean of the A gate is constant and represents the product of the average count rate and the gate width. It is therefore truly a background. For higher moments of the A gate, this is not the case, so that a simple subtraction cannot be done. It seems, therefore, that a multidimensional Rossi-α analysis, such as done by Bruggeman et al. [15, 25] may actually be needed.

## 4. Conclusion

This paper has discussed and compared joint-probability analyses and conditional-probability analyses. It has been shown in an intuitive way how they are related.

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# Acquisition Path Analysis Based on Material Flow Directed Graph Methodology

**Árpád Vincze,**

Hungarian Atomic Energy Authority
P.O.B.: 676, H-1539, Budapest, Hungary
E-mail: vincze@haea.gov.hu

**Anett Lukács**

Budapest University of Technology and Economics

**András Németh**

ESRI Hungary Ltd.

**Ákos Pető**

International Atomic Energy Agency

***Abstract:***

*The "directed graph analysis" has been shown to be a promising methodology to implement acquisition path analysis by the IAEA to support the State evaluation process [1]. Based on this methodology a material flow network model has been developed under the Hungarian Support Programme to the IAEA, in which materials in different chemical and physical form can flow through pipes representing declared processes, material transports, diversions or undeclared processes. The nodes of the network are the material types, while the edges of the network are the pipes [2]. The ranking of the resulting acquisition paths of the analysis is determined by the attributes of the processes (edges) included into the graph and the characteristics of the State. In this paper different set of attributes (cost, processing time, technical difficulty, free capacity, detectability, etc.) and their functional combination will be recommended for different analysis goals such as time, cost, technical difficulty or stealthiness of acquisition. Results will be presented for hypothetical case studies.*

**Keywords:** acquisition path analysis; directed graph; State level approach; material flow model;

## 1. Introduction

The successful implementation of an effective and efficient State-level safeguards approach requires that all plausible acquisition paths in a state are covered, applying various levels of safeguards measures and efforts tailored for a given state. This implies the differentiation of the states based on state-specific factors, the use of which must be well defined, as objective as possible and transparent. For finding the plausible acquisition paths in a given state, acquisition path analysis (APA) must be performed resulting in the clear understanding of the overall nuclear profile and proliferation potential of that state.

The "directed graph analysis" has been shown to be a promising methodology to implement acquisition path analysis to support the State evaluation process [1]. Based on this methodology a material flow network model has been developed under the Hungarian Support Programme to the IAEA, in which materials in different chemical and physical form can flow through pipes representing declared processes, material transports, diversions or undeclared processes.

Following the demonstration that the methodology can be successfully applied for the analysis [2], it was concluded that the key tasks for developing any acquisition path analysis methodology for IAEA safeguards purposes would have required the (i) determination of the set of material types and technological processes that describe the current technological possibilities relevant to nuclear weapons acquisition; (ii) the identification of all possible routes between materials, through processes that current technology permits; (iii) the identification and evaluation of the attributes and parameters to be used for the materials and processes of the graph model and (iv) the definition of the required analytical goals (weight functions) and corresponding mathematical algorithms [2]. This paper reports on a possible solution for these tasks.

## 2. The Material flow network model

The proposed model represents the nuclear fuel cycle (NFC) of a State by modelling the theoretical performance capacity of its nuclear facilities (including hypothetical clandestine facilities as well) by the possible ways of producing nuclear materials suitable for manufacturing a weapon or other nuclear explosive device. This approach is formulated via a directed graph (material flow network) model [1,2].

Materials in different chemical and physical forms can flow through pipes representing actual processes in the NFC. Any material flowing through the network can only change its state and quantity by passing through one of the pipes. A given pipe corresponds to an actual process in a given facility type having specific input and output material type. Transport, import and export of materials can also be taken into account by pipes. Each node of the network represents a material type of a given chemical and physical form, while the edges represent the processes (pipes).

Ranking of the possible paths can in principle be based on any function of the attributes assigned to the processes. The attributes (process parameters) of the processes used in our approach are summarized in Table 1.

Since the primary goal of the APA is to find all possible acquisition paths and provide with their ranking of plausibility, the parameters used in our analysis are relative based on a classification scheme as explained below.

Some of the process parameters shown in Table 1 can be estimated based on current state of art of nuclear technology knowledge, independently of the state where they are applied. These parameters (technical difficulty, maximum capacity, setup time and setup cost, processing time and processing cost of 1 significant quantity (SQ) of material, stealthiness) can therefore be estimated a priory for all possible technological processes that are considered by the IAEA using its technical knowledge base. The values used in this article for the case study are based on the expert judgement of the HAEA only and might need to be updated by a larger technical community.

There are three state-specific parameter shown in Table 1 (probability, the technical expertise of a state in relation to a given process and the declared flow). The declared flow of a process is given for declared processes by the declaration and zero for undeclared processes. The other two parameters must be estimated by expert judgment based on all available information about a state. The probability (*p*) is characterising the existence of a given process in the state having value 1 for declared processes and 0.01, 0.1 or 0.9 for undeclared processes of low, medium and high probability of existence, respectively. The state's technical expertise could be judged based on historical experience, general technological development of the state, scientific and technological publications and other open source information.

Using the process attributes given in Tabel 1, the following process length functions are defined:

Length-time:

$$L_{time} = t_p + t_s \cdot (1 - p) \cdot \frac{t_d}{t_e} \qquad \text{Eq.1.}$$

Length-cost:

$$L_{cost} = c_p + c_s \cdot (1 - p) \cdot \frac{t_d}{t_e} \quad \text{Eq.2.}$$

Length-misuse:

$$L_{misuse} = \left(1 - 10^{(-2 \cdot f / c_{max})}\right) \quad \text{Eq.3.}$$

Overall process length:

$$L_{all} = st + L_{time} + L_{cost} + L_{misuse} \quad \text{Eq.4.}$$

| **Process parameter** | **Description** | **Type** | **Base of classification** | **Value** |
|---|---|---|---|---|
| Probability -p | the probability of existence of a process | undeclared | low, medium, high | 0.01, 0.1, 0.9 |
| | | declared | declaration | 1 |
| Technical difficulty - $t_d$ | complexity of the process | all | low, medium, high | 0.01, 0.1, 1 |
| States technical expertise - $t_e$ | states former experience, industry capability | all | low, medium, high | 0.01, 0.1, 1 |
| Processing time – $t_p$ | time needed to process 1 SQ material | all | t< 3 month<br>3 month < t< 12 month<br>t > 12 month | 0.01<br>0.1,<br>1 |
| Setup time – $t_s$ | time to set up a process | undeclared | short, medium, long | 0.01, 0.1, 1 |
| cost - $c_s$ | cost of setting up the process for 1 SQ material | undeclared | low, medium, high | 0.01, 0.1, 1 |
| cost – $c_p$ | cost of processing 1 SQ material | all | low, medium, high | 0.01, 0.1, 1 |
| maximum capacity - $c_{max}$ | amount in SQ/year | declared<br>undeclared | declaration | as declared<br>1 |
| declared flow - f | declared material flow in SQ/year | declared<br>undeclared | declaration<br>- | as declared<br>0 |
| stealthiness - st | | declared<br>undeclared | declared<br>easy, medium, difficult | 1<br>0.01,0.1,1 |

**Table 1.** Process parameters.

The process cost and time length functions ($L_{time}$and $L_{cost}$, respectively) are accounting for the actual processing time and cost of 1 SQ material by a given process and also for the time and cost needed for setting up a process. This later value is only relevant for undeclared processes, and it is a function of the probability of existence of an assumed process (the lower the probability the more time and cost is foreseen to have the process in place) and also depends on the ratio of the technical difficulty versus the state's technical expertise. If the technical difficulty is high (1 for instance for gas centrifuge

enrichment) and the technical expertise is low (0.01), the function will return a high cost/time value, representing a long and costly process.

The misuse length function ($L_{misuse}$) enables the analysis to account for the potential misuse of a declared process by comparing the declared flow to the maximum theoretical flow as given in the DIQ of the facility in which the process is embedded. According to Eq. 3., if f = $c_{max}$, then the function returns 0.99 (there is low probability for misuse since the process is used with maximum declared capacity), if f=0 however, then $L_{misuse} = 0$, indicating that the declared process is not in declared use, so it is fully available for misuse.

The overall process length ($L_{all}$) is calculated by the sum of the time, cost, misuse process lengths and the stealthiness parameter, *st*.

As was explained in our previous paper, finding the possible shortest path between two nodes is carried out using Dijkstra's algorithm, in which the length of a path is the some of the length of each process along the path. The analyst can set any process attributes or one of the functions defined above as the process length determining parameter (i.e. the analyst can choose from which perspective the analysis is to be performed. The algorithm always finds the shortest path between two nodes once the length function is chosen. The default is to use the overall length function given by Eq.4.

The methodology has been implemented in the Esri®ArcMap™10.1 environment [3]. This software solution is a geospatial software application, based on ESRI's ArcGIS background. ArcGIS is a platform for building a complete geographic information system. This integrated collection of products lets us easily author data, diagrams, charts, maps, and globes on the desktop; publish them to a GIS server and/or share them online; and use them on the desktop, on the Web, or in the field. With this background our software solution provides a standards-based tool for data management, spatial analysis, and mapping.

The implementation of the methodology for a given state requires the following steps:

- Defining materials and processes based on the declared fuel cycle of the State and declared material flows for a given year;
- Making hypothesis on the presence of any undeclared materials and processes in the State including diversion and misuse scenarios;
- Determining process parameters as described above;
- Connecting materials and processes following their connectivity rules (making the graph of the State);
- Setting process length function (type of analysis);
- Setting starting and end material types;
- Finding the shortest acquisition path;
- Finding all acquisition paths between the two nodes sequentially by blocking the previous shorter path;

By mapping out all possible source and end materials (HEU, or Pu metal), the analysis provides with a ranked list of all the acquisition paths of the particular state. The ranking order might change with the change of the process length function (timeliness, cost, stealthiness, misuse analysis, or all combined), but the total list is always the same.

The actual performance of the methodology will be illustrated below using information of a hypothetical state as a case study.

### 3. Case study – Information of the Hypo state

A hypothetical state (named here as Hypo) information will be used as a case study. Hypo as a non-nuclear weapon State, was an early party to the Treaty on the Non Proliferation of Nuclear Weapons (NPT), and followed with the conclusion of the comprehensive safeguards agreement (CSA), which is in force since 1976. Hypo also signed the Additional Protocol (AP) recently, but has not yet ratified it. In addition, Hypo is a party or is active as a member of the Chemical Weapons Convention (CWC), the Comprehensive Test Ban Treaty (CTBT) and bilateral agreements with all trading partners.

Hypo has large uranium deposits in several areas in the country and developed its nuclear fuel cycle to produce 20% of its electricity needs. Hypo has significant activities in several stages of the fuel cycle, and remains committed to its nuclear program. The declared facilities are listed in Table 2.

Nuclear fuel cycle activities are taking place in two areas: Area 1 and Area 2. There are two uranium mines and one concentration plant in Area 1. The entire product (yellow cake) of the concentration plant is exported. Major part of the exported material is shipped to Neighbour1 for being enriched in isotope U235. The low enriched uranium in the form of $UO_2$ powder is shipped back to Hypo in order to be used as a feed to the Fuel Fabrication Plant which manufactures LWR fuel assemblies for domestic use. As current mining and concentration capacities provide uranium amounts covering only about 50% of Hypo's needs, it plans to develop additional uranium production capacities.

There are two NPP stations (NPPs) in Area 2: NPP1, consisting of two reactor blocks, and NPP2, consisting of four reactor blocks. Each block represents a standard pressurized light water reactor (1000 MW(e)). The reactors were imported from Neigbour2. There is a separate long-term dry storage facility in Area 2 for spent fuel coming from all the reactors. Nuclear fuel for the NPPs is manufactured in the Fuel Fabrication Plant (FFP) which is also located in Area 2. This FFP was purchased, as a turn-key plant, from Neighbour2. Feed material for the NPP, the $UO_2$ powder with enrichment in isotope U235 from 3% to 4%, is also received from Neighbour2.

Over the next ten years, Hypo plans further development of its NFC activities. There are plans for further development related to the production of source material (new underground mine and development of a uranium in-situ leaching mine, to complete the construction of new concentration plant). Hypo is exploring the possibility of purchasing a turn-key centrifuge enrichment plant from Neighbour2 in order to be self-sufficient in the production of fuel for its power reactors. Hypo plans construction of a new nuclear power station comprising two 1000 MW(e) PWR reactors in Area 2. Hypo also considers developing their nuclear industrial capabilities to have at least 50% of their NPP future program developed on indigenous capabilities. Hypo is in the process of determining whether they wish to pursue reprocessing of their spent reactor fuel. They might consider starting R&D work on reprocessing. However, no decision has been taken so far.

| NFC level | Facility | Annual capacity | Production or inventroy in 2011 |
|---|---|---|---|
| Mining | Area 1 - South | 150 000 tons of U ore | 150 000 tons U ore |
| | Area 1 – North (temporarily) | 100 000 tons of U ore | 100 000 tons of U ore |
| Milling (Concentration) | Area 1 | 300 tons UOC (U3O8) | 290 tons |
| Export | Area 1 | - | 285.6 tons |
| Import | Area 2 | - | 120 tons of UO2 LEU |
| LEU Fuel Fabrication | Area 2 - FFP | 300 LWR fuel (400 kg uranium 3.5 % enriched per fuel) | 300 LWR (throughput 120 t of LEU) |
| Reactors<br>Six PWR – 1000 MW(e)<br>Each reactor unit includes fresh fuel storage and a cooling pool for spent fuel. | Area 2<br>NPP1: NPPA, NPPB,<br>NPP2: NPPC, NPPD, NPPE, NPPF | FF: 100 fresh fuel<br>CF: 150 assemblies, 1/3 refuelled<br>SF: 1200 spent fuel | NPP FF CF SF<br>A 10 150 300<br>B 50 0 500<br>C 0 150 300<br>D 70 150 250<br>E 10 150 300<br>F 10 150 250 |
| SF dry storage facility | Area 2 | 20 modules<br>60 SF per module | 180 SF assemblies |

**Table 2.** Hypo's nuclear fuel cycle (NFC) facilities and inventories

## 4. Process parameters and analysis input

Based on the information available about Hypo, one can immediately see that the nuclear fuel cycle of the State does not contain declared processes the misuse of which could result in the production of weapon usable materials. For the acquisition of direct use materials therefore, Hypo must have some

undeclared activities. For the analysis we assume with low probability that Hypo has conversion capabilities, already has experiencing with the gas-centrifuge enrichment technology and developed the technology needed for dissolution of fuel assemblies and/or irradiated targets and separation of uranium and plutonium. In addition it is assumed that covert mining and milling also present in the country to have larger quantity of source material. These activities are assumed because Hypo's future plans cover these technologies anyway. The hypothesis of low probability is justified by the fact that Hypo signed the Additional Protocol and also shows commitment to other non-proliferation instruments.

Based on these assumptions and the data shown in Table 2, the type of processes and their attributes are listed in Table 3. Processes of the same type and same attributes corresponding to the same facility (i.e. NPP1, NPP2 reactors) are only shown in one row with the number of identical processes indicated in brackets. Undeclared processes are marked with gray background.

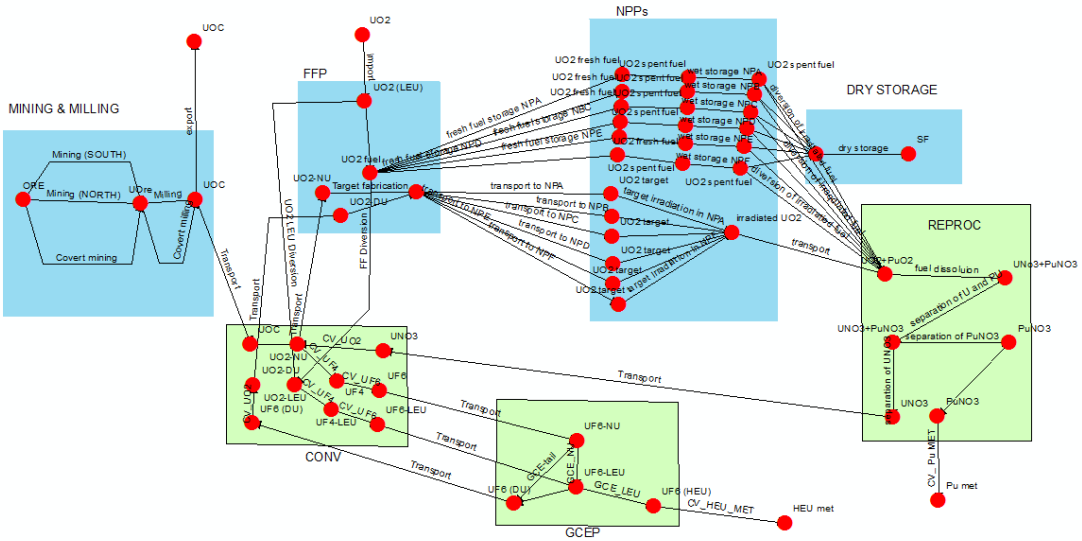


**Figure 1.** The directed graph representing Hypo state's material processing capabilities.

The directed graph build from the processes in Table 3 is shown in Figure 1. The processes can be assigned to facility types as shown in the Figure with blue and green rectangles representing declared and undeclared facilities, respectively.

Altogether 18 material types and 82 different processes has been defined for Hypo, including import, export and transport of materials between facilities (see Table 2).

The analysis is done using ArcMap's built in Dijkstra's algorithm, where the starting and end nodes can be graphically selected. After the shortest path is found, an alternate path is found by blocking one of the processes at junctions along the path. This tool can also be used to model the efficiency of safeguarding a given process as far as the reduction of the number of acquisition paths are concerned, thereby making optimisation of the safeguards implementation plan for a state possible.

## 5. Analysis results

The results of the acquisition path analysis for Hypo based and the directed graph shown in Figure 1 and the input parameters listed in Table 3, are ranked in Table 4. The ranking is based on the $L_{all}$ process length parameter. Paths with exactly the same characteristics are listed only in one row indicating the total number of similar processes in the last column of Table 4. Similar paths are due to several similar types of processes representing alternate directions along a path (i.e. the same activities can be done in all the six reactors).

| NFC_level | Process name | Material From-To | Probability (p) | Technical difficulty (td) | State's technical expertise (te) | Time, process (tp) | Time to set-up (ts) | Cost of processing (cp) | Cost to set-up (cs) | Maximum capacity (cmax) [SQ/year] | Declared flow (f) | Stealthiness (st) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| MINING & MILLING | Mining (NORTH) | ORE-ORE | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.1000 | 0.0100 | 0.0100 | 14.0000 | 14.0000 | 1.0000 |
| | Mining (SOUTH) | ORE-ORE | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.1000 | 0.0100 | 0.0100 | 14.0000 | 14.0000 | 1.0000 |
| | Milling | ORE-UOC | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.1000 | 0.0100 | 0.0100 | 13.8000 | 13.3000 | 0.1000 |
| | Export | UOC-UOC | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 13.3000 | 13.1000 | 0.0100 |
| | Covert mining | ORE-ORE | 0.1000 | 0.0100 | 1.0000 | 0.0100 | 0.1000 | 0.0100 | 0.0100 | 1.0000 | 0.0000 | 0.0100 |
| | Covert milling | ORE-UOC | 0.1000 | 0.0100 | 1.0000 | 0.0100 | 0.1000 | 0.0100 | 0.0100 | 1.0000 | 0.0000 | 0.0100 |
| | Transport UOC | UOC-UOC | 0.0100 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 1.0000 | 0.0000 | 0.0100 |
| FUEL FABRICATION | import | UO2LEU-UO2LEU | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 32.0000 | 32.0000 | 0.0100 |
| | fuel fabrication | UO2LEU-FF | 1.0000 | 0.1000 | 1.0000 | 0.1000 | 0.1000 | 0.0100 | 0.1000 | 64.0000 | 64.0000 | 1.0000 |
| | Target fabrication-NU | UO2NU-UO2TNU | 0.0100 | 0.1000 | 1.0000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 0.0100 |
| | Target fabrication-DU | UO2DU-UO2TDU | 0.0100 | 0.1000 | 1.0000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 0.0100 |
| | UO2 LEU Diversion | UO2LEU-UO2LEU | 0.0100 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 1.0000 | 0.0000 | 0.1000 |
| | FF Diversion | FF-FF | 0.0100 | 0.1000 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 1.0000 | 0.0000 | 0.1000 |
| | Transport | UO2NU-UO2NU | 0.0100 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 1.0000 | 0.0000 | 0.0100 |
| | Transport | UO2DU-UO2DU | 0.0100 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 1.0000 | 0.0000 | 0.0100 |
| NPPs | fresh fuel storage NPA | FF-FF | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 21.3000 | 2.1000 | 1.0000 |
| | fresh fuel storage NPB | FF-FF | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 21.3000 | 10.7000 | 1.0000 |
| | fresh fuel storage NBC | FF-FF | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 21.3000 | 0.0000 | 1.0000 |
| | fresh fuel storage NPD | FF-FF | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 21.3000 | 14.9000 | 1.0000 |
| | fresh fuel storage NPE | FF-FF | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 21.3000 | 2.1000 | 1.0000 |
| | fresh fuel storage NPF | FF-FF | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 21.3000 | 2.1000 | 1.0000 |
| | fuel irradiation NPA | FF-SF | 1.0000 | 0.1000 | 1.0000 | 0.0100 | 0.1000 | 0.0100 | 1.0000 | 32.0000 | 32.0000 | 1.0000 |
| | fuel irradiation NPB | FF-SF | 1.0000 | 0.1000 | 1.0000 | 0.0100 | 0.1000 | 0.0100 | 1.0000 | 32.0000 | 0.0000 | 1.0000 |
| | fuel irradiation NPC | FF-SF | 1.0000 | 0.1000 | 1.0000 | 0.0100 | 0.1000 | 0.0100 | 0.0100 | 32.0000 | 32.0000 | 1.0000 |
| | fuel irradiation NPD | FF-SF | 1.0000 | 0.1000 | 1.0000 | 0.0100 | 0.1000 | 0.0100 | 0.0100 | 32.0000 | 32.0000 | 1.0000 |
| | fuel irradiation NPE | FF-SF | 1.0000 | 0.1000 | 1.0000 | 0.0100 | 0.1000 | 0.0100 | 0.0100 | 32.0000 | 32.0000 | 1.0000 |
| | fuel irradiation NPF | FF-SF | 1.0000 | 0.1000 | 1.0000 | 0.0100 | 0.1000 | 0.0100 | 0.0100 | 32.0000 | 32.0000 | 1.0000 |
| | wet storage NPA | SF-SF | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 256.0000 | 64.0000 | 1.0000 |
| | wet storage NPB | SF-SF | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 256.0000 | 106.7000 | 1.0000 |
| | wet storage NPC | SF-SF | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 256.0000 | 64.0000 | 1.0000 |
| | wet storage NPD | SF-SF | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 256.0000 | 53.3000 | 1.0000 |

| | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | wet storage NPE | SF-SF | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 256.0000 | 64.0000 | 1.0000 |
| | wet storage NPF | SF-SF | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 256.0000 | 53.3000 | 1.0000 |
| | SF diversion at NPX (6) | SF-SF | 0.0100 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 1.0000 | 0.0000 | 0.1000 |
| | SF Transport | SF-SF | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 30.0000 | 30.0000 | 1.0000 |
| | Target transport to NPX (6) | Target- Target | 0.0100 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 1.0000 | 0.0000 | 0.0100 |
| | target irradiation in NPX (6) | Target –I.Target | 0.0100 | 0.1000 | 1.0000 | 0.1000 | 0.1000 | 0.0100 | 0.1000 | 1.0000 | 0.0000 | 0.1000 |
| | Irradiated target transport | I.Target- I.Target | 0.0100 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 1.0000 | 0.0000 | 0.0100 |
| REPROCESSING | fuel dissoluion | SF – SF sol. | 0.0100 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 1.0000 |
| | separation of U and PU | SF sol.-UNO3,PuNO3 | 0.0100 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 0.1000 |
| | separation of UNO3 | UNO,PuNO3-UNO3 | 0.0100 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 0.1000 |
| | Transport UNO3 | UNO3-UNO3 | 0.0100 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.1000 | 0.0100 | 1.0000 | 0.0000 | 0.0100 |
| | separation of PuNO3 | UNO,PuNO3-PuNO3 | 0.0100 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 0.1000 |
| | purification of PuNO3 | PuNO3-PuNO3 | 0.0100 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 0.1000 |
| CONVERSION | CV PuNO3 to Pu MET | PuNO3-PuMET | 0.0100 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 0.0100 |
| SF STORAGE | dry storage | SF-SF | 1.0000 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 76.8000 | 38.4000 | 1.0000 |
| | irradiated fuel diversion | SF-SF | 0.0100 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 1.0000 | 0.0000 | 0.1000 |
| CONVERSION | CV_UO2 | UOC-UO2NU | 0.0100 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 0.0100 |
| | CV_UO2 | UOF6DU- UO2DU | 0.0100 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 0.0100 |
| | CV_UO2 | UNO3- UO2 | 0.0100 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 0.0100 |
| | CV_UF4 | UO2LEU-UF4LEU | 0.0100 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 0.1000 |
| | CV_UF6 | UF4LEU- UF6LEU | 0.0100 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 0.1000 |
| | CV_UF4 | UO2NU-UF4NU | 0.0100 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 0.1000 |
| | CV_UF6 | UF4NU- UF6NU | 0.0100 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 0.1000 |
| | CV_HEU_MET | UF6HEU -HEUMET | 0.0100 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 0.1000 | 1.0000 | 0.0000 | 0.0100 |
| GCEP | GCE_LEU | UF6LEU- UF6HEU | 0.0100 | 1.0000 | 0.1000 | 0.1000 | 1.0000 | 0.0100 | 1.0000 | 1.0000 | 0.0000 | 0.1000 |
| | GCE_NU | UF6NU- UF6LEU | 0.0100 | 1.0000 | 0.1000 | 1.0000 | 1.0000 | 0.1000 | 1.0000 | 1.0000 | 0.0000 | 0.1000 |
| | GCE-tail | UF6LEU- UF6DU | 0.0100 | 1.0000 | 0.1000 | 1.0000 | 1.0000 | 0.1000 | 1.0000 | 1.0000 | 0.0000 | 0.1000 |
| | GCE-tail | UF6NU- UF6DU | 0.0100 | 1.0000 | 0.1000 | 1.0000 | 1.0000 | 0.1000 | 1.0000 | 1.0000 | 0.0000 | 0.1000 |
| | Transport-NU | UF6NU- UF6NU | 0.0100 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 1.0000 | 0.0000 | 0.0100 |
| | Transport - DU | UF6DU- UF6DU | 0.0100 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 1.0000 | 0.0000 | 0.0100 |
| | Transport - LEU | UF6LEU- UF6LEU | 0.0100 | 0.0100 | 1.0000 | 0.0100 | 0.0100 | 0.0100 | 0.0100 | 1.0000 | 0.0000 | 0.0100 |

**Table 3.** Processes and process attributes defined for Hypo state

| Paths_ID | From | To | N-steps | L-value-all | L-time | L-cost | st | misuse | N-paths |
|---|---|---|---|---|---|---|---|---|---|
| 51 | ORE | PU-MET | 14 | 4.350372 | 1.475996 | 1.384376 | 1.49 | 0 | 6 |
| 57 | ORE | PU-MET | 14 | 6.327566 | 1.475096 | 1.384286 | 2.48 | 0.988184 | 6 |
| 63 | ORE | PU-MET | 14 | 6.329382 | 1.475096 | 1.384286 | 2.48 | 0.99 | 6 |
| 81 | ORE | PU-MET | 14 | 6.329382 | 1.475096 | 1.384286 | 2.48 | 0.99 | 6 |
| 69 | ORE | PU-MET | 14 | 8.306576 | 1.474196 | 1.384196 | 3.47 | 1.978184 | 6 |
| 75 | ORE | PU-MET | 14 | 8.306576 | 1.474196 | 1.384196 | 3.47 | 1.978184 | 6 |
| 123 | UO2-LEU | PU-MET | 11 | 12.26397 | 1.145099 | 1.055099 | 6.41 | 3.653772 | 1 |
| 124 | UO2-LEU | PU-MET | 11 | 12.34458 | 1.145099 | 1.055099 | 6.41 | 3.734383 | 1 |
| 125 | UO2-LEU | PU-MET | 11 | 12.56179 | 1.145099 | 1.055099 | 6.41 | 3.951588 | 1 |
| 126 | UO2-LEU | PU-MET | 11 | 12.62891 | 1.145099 | 1.055099 | 6.41 | 4.018709 | 1 |
| 127 | UO2-LEU | PU-MET | 11 | 12.62891 | 1.145099 | 1.055099 | 6.41 | 4.018709 | 1 |
| 128 | UO2-LEU | PU-MET | 11 | 13.15695 | 1.145099 | 1.055099 | 6.41 | 4.546755 | 1 |
| 136 | UO2-LEU | PU-MET | 12 | 13.28397 | 1.155099 | 1.065099 | 6.42 | 4.643772 | 1 |
| 137 | UO2-LEU | PU-MET | 12 | 13.36458 | 1.155099 | 1.065099 | 6.42 | 4.724383 | 1 |
| 138 | UO2-LEU | PU-MET | 12 | 13.58179 | 1.155099 | 1.065099 | 6.42 | 4.941588 | 1 |
| 139 | UO2-LEU | PU-MET | 12 | 13.64891 | 1.155099 | 1.065099 | 6.42 | 5.008709 | 1 |
| 140 | UO2-LEU | PU-MET | 12 | 13.64891 | 1.155099 | 1.065099 | 6.42 | 5.008709 | 1 |
| 141 | UO2-LEU | PU-MET | 12 | 14.17695 | 1.155099 | 1.065099 | 6.42 | 5.536755 | 1 |
| 7 | UO2-LEU | HEU | 7 | 23.5744 | 10.6272 | 10.5372 | 1.42 | 0.99 | 1 |
| 8 | UO2-LEU | HEU | 8 | 24.67618 | 10.72809 | 10.54809 | 1.42 | 1.98 | 1 |
| 87 | ORE | PU-MET | 20 | 26.81477 | 12.99319 | 12.00157 | 1.82 | 0 | 6 |
| 99 | ORE | PU-MET | 20 | 27.80378 | 12.99229 | 12.00148 | 2.81 | 0 | 6 |
| 111 | ORE | PU-MET | 20 | 27.80378 | 12.99229 | 12.00148 | 2.81 | 0 | 6 |
| 129 | UO2-LEU | PU-MET | 18 | 28.44479 | 12.78239 | 11.79239 | 2.88 | 0.99 | 1 |
| 93 | ORE | PU-MET | 20 | 28.79196 | 12.99229 | 12.00148 | 2.81 | 0.988184 | 6 |
| 105 | ORE | PU-MET | 20 | 28.79279 | 12.99139 | 12.00139 | 3.8 | 0 | 6 |
| 117 | ORE | PU-MET | 20 | 28.79279 | 12.99139 | 12.00139 | 3.8 | 0 | 6 |
| 130 | UO2-LEU | PU-MET | 19 | 30.54657 | 12.88329 | 11.80329 | 3.88 | 1.98 | 6 |
| 1 | ORE | HEU | 10 | 42.94438 | 21.738 | 20.74638 | 0.46 | 0 | 1 |
| 4 | ORE | HEU | 10 | 44.92157 | 21.7371 | 20.74629 | 1.45 | 0.988184 | 1 |
| 2 | ORE | HEU | 10 | 44.92339 | 21.7371 | 20.74629 | 1.45 | 0.99 | 1 |
| 3 | ORE | HEU | 10 | 44.92339 | 21.73709 | 20.74629 | 1.45 | 0.990008 | 1 |
| 15 | ORE | HEU | 20 | 46.41677 | 22.79419 | 21.80257 | 1.82 | 0 | 6 |
| 5 | ORE | HEU | 10 | 46.90058 | 21.7362 | 20.7462 | 2.44 | 1.978184 | 1 |
| 6 | ORE | HEU | 10 | 46.90058 | 21.7362 | 20.7462 | 2.44 | 1.978184 | 1 |
| 33 | ORE | HEU | 20 | 47.49396 | 22.79329 | 21.80248 | 1.91 | 0.988184 | 6 |
| 21 | ORE | HEU | 20 | 48.39578 | 22.79329 | 21.80248 | 2.81 | 0.99 | 6 |
| 27 | ORE | HEU | 20 | 48.39578 | 22.79329 | 21.80248 | 2.81 | 0.99 | 6 |
| 39 | ORE | HEU | 20 | 50.37297 | 22.79239 | 21.80239 | 3.8 | 1.978184 | 6 |
| 45 | ORE | HEU | 20 | 50.37297 | 22.79239 | 21.80239 | 3.8 | 1.978184 | 6 |
| 9 | UO2-LEU | HEU | 17 | 53.34037 | 22.4633 | 21.4733 | 5.75 | 3.653772 | 1 |
| 10 | UO2-LEU | HEU | 17 | 53.42098 | 22.4633 | 21.4733 | 5.75 | 3.734383 | 1 |
| 11 | UO2-LEU | HEU | 17 | 53.63818 | 22.4633 | 21.4733 | 5.75 | 3.951588 | 1 |
| 12 | UO2-LEU | HEU | 17 | 53.7053 | 22.4633 | 21.4733 | 5.75 | 4.018709 | 1 |
| 13 | UO2-LEU | HEU | 17 | 53.7053 | 22.4633 | 21.4733 | 5.75 | 4.018709 | 1 |
| 14 | UO2-LEU | HEU | 17 | 54.23335 | 22.4633 | 21.4733 | 5.75 | 4.546755 | 1 |
| | | | | | | | | Total number: | 141 |

**Table 4.** Ranked list of the possible acquisition paths found for Hypo state based on the $L_{all}$ parameter.

It can be seen from Table 4, that in total 141 different acquisition paths could be found for Hypo state, the ranking of which is based on the choice of the process length parameter. If all the four aspects (time, cost, stealthiness and misuse) are taken into account, the relative rank is as shown in Table 4.

The first row of Table 4 represents the shortest six paths with identical path length (i.e. the most possible ones) that are starting from uranium ore and leading to plutonium metal as shown with light blue lines in Figure 2. These paths are comprised of undeclared processes only via the following steps: covert mining->covert milling->transport of UOC to undeclared conversion facility (CONV)->UOC conversion to $UO_2$->transport $UO_2$ to Fuel Fabrication Facility->undeclared target fabrication->transfer to one of the reactors->target irradiation->transfer of irradiated target to undeclared reprocessing->fuel dissolution->$UNO_3$-$PuNO_3$ separation->$PuNO_3$ separation->$PuNO_3$ purification->conversion to Pu metal.

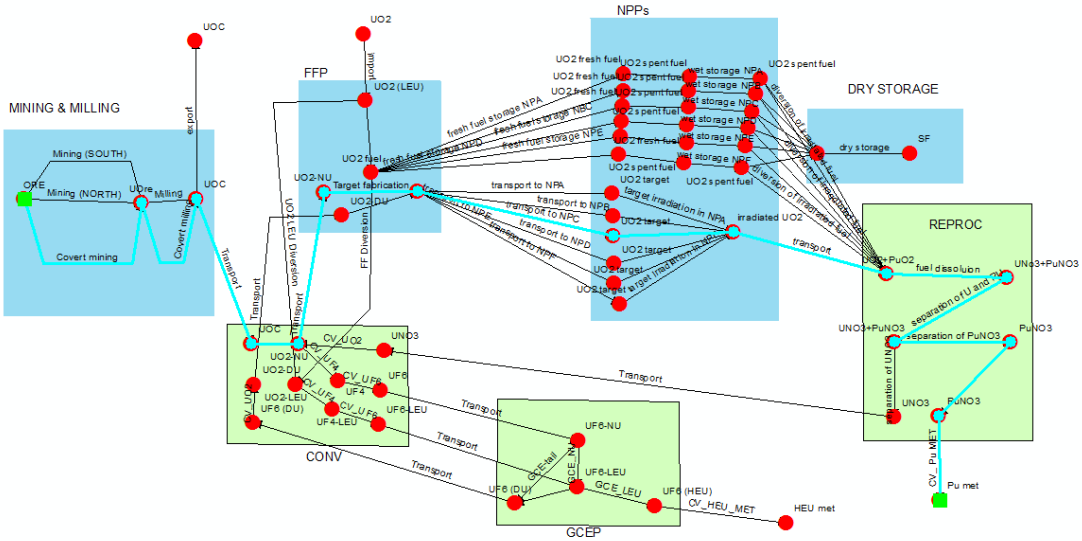


**Figure 2.** One possible acquisition path for Hypo state starting from uranium ore (ORE) and leading to plutonium metal (Pu Met).

It follows from Table 4 that these paths are, however, not the ones comprising of the least number of steps (14 steps as compared to a 7 step path leading to HEU), nor has the lowest time, cost and stealthiness parameters, but the sum of all the four parameters is the lowest possible. This illustrates well that the relative order of the possible paths depends on the process length function set for the analysis.

It should also be noticed from Table 4, that the paths leading to plutonium (Pu Met) are in general have lower path length values that the ones leading to highly enriched uranium (HEU), despite the fact that for both the undeclared reprocessing and gas centrifuge enrichment was assumed to be present in Hypo with the same low probability (see Table 3). The difference in path length is due to the fact that the technical difficulty of GCE is set higher than that of the reprocessing, while the Hypo's technical expertise is intermediate in both cases. This results in higher cost and time length values for paths having GCE processes as compared to those without. This well illustrates the importance of the state specific parameters in the methodology as far as the prioritisation of the possible acquisition paths is concerned.

## 6. Conclusion

The main points of the present paper are as follows:

(i) For states with open nuclear fuel cycles acquisition of weapon grade nuclear material is not possible without the assumption of the presence of undeclared activities in the State. According to the state-level safeguards approach, this assumption should be based on all available information about the State. One way to do this is to apply the hypothesis that was used to establish the Safeguards Criteria: assume that all clandestine activities

needed to further process any diverted materials are present in the State irrespective of any other circumstances. It was pointed out however, that this approach leads to a mechanistic, rigid and inefficient facility level safeguards system that can universally be applied to any State with comprehensive safeguards agreement in force [1]. It is suggested therefore that a more realistic hypothesis about a state's theoretical capabilities should be made and continuously challenged via state-level safeguards measures and updated regularly based on new information gathered [2].

(ii) The number of acquisition paths is determined by the assumptions of the hypothesis about a state and is exponentially grows with the number of undeclared processes considered to be present in the state. The application of the universal hypothesis mentioned in the previous paragraph leads therefore to virtually unlimited number of possible acquisition paths even for States with very limited declared nuclear activities. The authors believe that more plausible assumptions could be made about a State, for which the IAEA should develop its own guidance (how to use all the information of a Sate?).

(iii) Since the primary goal of the APA is to find all possible acquisition paths and provide with their prioritisation rank, in our methodology relative parametric values are used to characterise processes based on a simple logarithmic classification scheme. Most of the process parameters could be realistically estimated based on the knowledge of the nuclear technology only. These parameters (technical difficulty, maximum capacity, setup time and setup cost, processing time and processing cost of 1 significant quantity (SQ) of material, stealthiness, etc.) can therefore be estimated a priory for all the possible technological processes that are accounted for by the IAEA using its technical knowledge base. This should form a solid common base of the acquisition path analysis for all states.

(iv) In addition, state-specific parameters have also been suggested such as the probability of the existence of a process in a state, the technical expertise of a state in relation to a given process and declared flow of processes, which makes the methodology capable to result in different prioritization of acquisition paths for States with exactly the same nuclear fuel cycle. These parameters should be estimated based on the State's declaration and by expert judgment based on historical experience, general technological development of the state, scientific and technological publications and other open source information.

(v) The present implementation of the methodology can be used for qualitative analysis resulting in the list of all possible acquisition paths in a given state. The priority ranking of this set of path is possible and the result depends on the selected aspect of the analysis. Parameter functions for process cost, process time, stealthiness, misuse have been formulated and tested in this paper for practical usage with promising results for the case of a hypothetical state.

(vi) Further work is necessary, however to select the most appropriate form of these parameter functions. However, that work must rely on the "user requirements" that could only be specified by the IAEA.

## 7. References

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# 8 Legal matters

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# From DCM14 to the Next Generation Surveillance System – Experience and Plans for Future Implementation

**K. Schoop, J. Pekkarinen, S. Pirvulet, J.-M. Mazur (1)**
**M. Moeslinger, G. Hadfi (2)**

(1) European Commission, DG Energy, Unit E1, L- 2920 Luxembourg
(2) International Atomic Energy Agency, SGTS, A-1400 Vienna
E-mail: Konrad.Schoop@ec.europa.eu

***Abstract:***

*With the introduction of the Integrated Safeguards concept the role of a highly reliable, state of the art surveillance system became more important. The implemented DCM14 based surveillance systems have shown a very good reliability in the field over several years, they are indispensable as our current workhorse but have some limitations concerning new requirements for example regarding the storage, cryptography and remote data transmission (RDT) capabilities.*
*The development of the Next Generation Surveillance Systems (NGSS) is nearly finished now and the field tests with the prototypes are showing very promising results. Considering the very large investment in surveillance systems, a prudent approach assuring the most efficient use of resources is required when changing from DCM14 to the NGSS. Existing systems will be exchanged step by step over many years. Due to the successful cooperation and information exchange with the IAEA, new ideas were born for further improvements of NGSS, so that the software upgrades will allow a joint use and cost effective application.*
*To allow extended use of the existing systems, we made successful tests to upgrade them in a cost effective manner and replace the DCM14 systems only near the end of the life time or where the upgrade to the NGSS is a prerequisite to allow the implementation of RDT. Due to the downwards compatibility of the NGSS components, a step by step migration is possible from the DCM14 to the NGSS technology. This process will take place in the next 5 to 10 years and will allow us to adapt the intensive replacement workload to the available financial and human resources.*
*An outlook for the future requirements for surveillance system replacements will be provided.*

**Keywords:** Next Generation Surveillance System (NGSS), high reliability, data transfer security

## 1. Introduction

The Direction Nuclear Safeguards of the DG Energy of the European Commission (further named as EURATOM) and the IAEA Safeguard Department (here named as IAEA) are commonly installing, operating, reviewing and maintaining about 350 cameras for safeguard purposes in the nuclear installation in the Non-Nuclear Weapon States of the European Union (EU-NNWS).

Most of these cameras are part of different surveillance systems based on the custom designed Digital Camera Module DCM-14 [1]. The modular design of the DCM-14 based surveillance systems allows a great flexibility and adaptation to the surveillance needs, from portable All-In-one-systems (ALIS), over single (DSOS) to different multiple camera systems (DMOS, SDIS) [2]. The DCM-14 is digitising the images from a low power CCTV camera with a selectable heartbeat, has different trigger capabilities like time lapse, scene change, seal, external and housing triggers with pre- and post-image selection and a selectable image compression to reduce the storage space. All triggered images are digitally signed using a symmetric key and the Triple-DES (Digital Encryption Standard) algorithm to ensure that the image is authentic. All images can also be encrypted using the DES algorithm. The images are stored on a commercial PC card which is either the storage media (for an ALIS) or a buffer of about 250 000 images for the other data recording media in the systems. The DCM-14 needs external power, either 100…240 VAC (for internal ALIS power supply) or 10…36 VDC coming from the recording system. The clever built-in power management together with a commercially available

backup battery allow an independent powering of the DCM14 and the camera for an image taking of up to 4500 images so that a power loss period up to 4 weeks can be covered. Also due to its exceptional reliability – the camera module is self-checking its own software and can recover from radiation induced SEU (Single Event Upsets [3]) - the DCM-14 is a very valuable safeguards tool and can be used for the next several years.

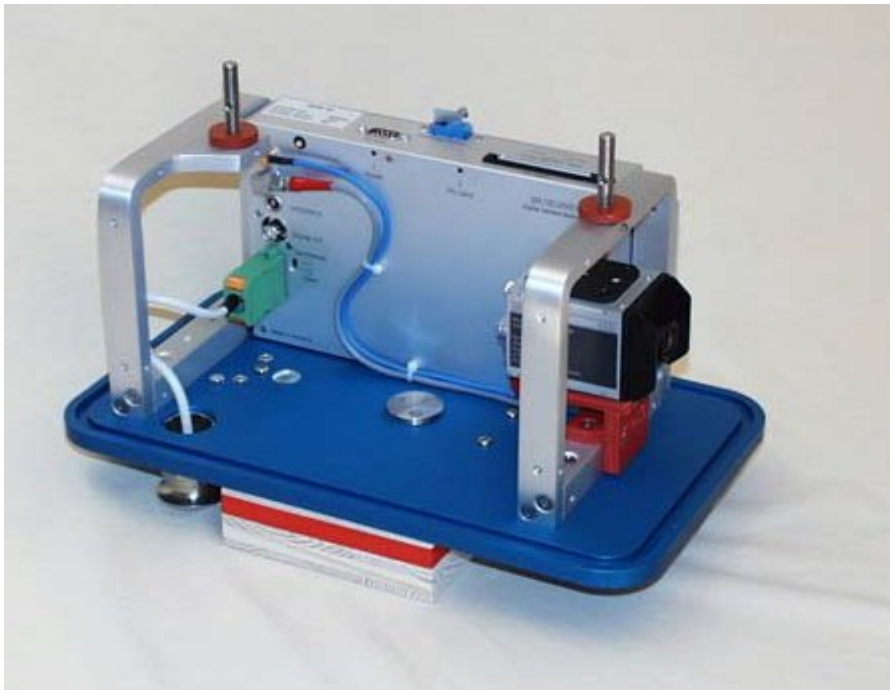

Figure 1: DCM-14 with a low power CCTV camera mounted in a camera housing (without cover)

But the DCM-14 was developed in the last millennium, the chip sets and development tools are obsolete and not available anymore, the RDT (Remote Data Transmission) capability is limited and has to be covered by an additional PC (running either on Windows NT or XP SP1), the authentication and encryption are using symmetric keys and the cryptographic algorithms might be not safe enough in a few years. So the IAEA started the development of the successor system with a market survey in 2005.

## 2. Development of the Next Generation Surveillance System (NGSS)

From the very beginning the IAEA and EURATOM jointly developed the NGSS user requirements. Based on the positive experience with the DCM-14, most features of this device were either maintained or improved, mainly the security and the RDT features. The result of the market survey was that such a device was not commercially available and there was no intention to be developed, because the security market does not have needs for sophisticated security features, long power loss operation, SEU resistance and radiation tolerance. So the IAEA was helped by their member states support programmes (MS-SP) to conduct the development of a new Safeguards surveillance technology. It was a joint development between the German (Ge-) and US-SP, whereas the Ge-SP was financing the development of the camera and camera interface and the US-SP the development of the data consolidater and the review software.

The NGSS system provides the complete surveillance infrastructure needed for imaging and equipment state-of-health data retrieval, extremely important for drawing of safeguards relevant conclusions [4]. Visual evidence of events is recorded and processed in a front-end camera and stored locally or forwarded to a data consolidator unit where data is stored and forwarded via a remote monitoring connection (where installed). At the back-end, the surveillance review software performs the analysis of image files with an automatic data filtering and pre-processing system and provides as well the tools for an efficient review by the safeguards inspectors. The whole NGSS system was designed for ease of use and maintenance with a modular infrastructure that allows a simpler inventory management, uncomplicated (plug-and-play) exchanges of faulty modules in the field, and easier upgrade as new technologies become available.

The NGSS hardware design evolved based on the need for a highly reliable and efficient surveillance system and on the lessons learned from prior system/cameras used by the IAEA and EURATOM. The key design features which set NGSS apart from prior systems and from currently available commercial systems are its data security, modularity and scalability, its compatibility with the existing surveillance infrastructure already installed in facilities and its standardization and sustainability. Newly available technologies were incorporated to significantly improve the equipment performance characteristics, reduce the total cost of ownership and simplify the implementation of NGSS.

NGSS was deliberately designed to be able to reuse already existing expensive facility infrastructure like camera communication lines and power cables. Additionally, the NGSS server hardware can also be connected to legacy cameras, allowing mixed configuration surveillance systems with both, new NGSS cameras and existing DCM-14 based legacy cameras. This backward compatibility allows a flexible, seamless and costs effective transition from the legacy systems to the NGSS.

During development the ideas from developers were incorporated and therefore the user requirements were improved in 2009. As a result the features of the NGSS system have at least the same or better performance as the DCM-14 i.e. shorter PTI (Picture Taking Interval) down to 1 second, longer backup battery autonomy until few months and bigger data storage capacity. Important features where big steps forward were made are mainly on the data security like authentication, encryption, remote access authorisation and component identification as well as in the tamper indication [5].

The authenticity and confidentiality of surveillance data acquired by NGSS is guaranteed by three different layers of asymmetric cryptographic data protection and two layers of physical, passive and active tamper indication. The innermost layer data authentication is provided by a digital cryptographic signature. All data is then encrypted before being stored on removable flash media inside the camera. As a third layer the communication of the camera via a network or serial connection is protected using Transaction Layer Security (TLS). Standard, well documented public key cryptographic methods and cipher suites are used for each layer. At the heart of the NGSS camera a secure Surveillance Core Component (SCC) protects the critical camera electronic components, the optical sensor as well as the cryptographic secrets with an advanced, active tamper indication mechanism (ATI). The NGSS camera is designed to be electronically sealed at all times and does therefore not need to rely on the protection of a sealable outer enclosure or housing.

The elimination of the requirement for an external housing seal and the support for public key data authentication allow that the NGSS can easily be jointly used with the other inspectorates or states with each party being independently able to confirm data authenticity. The NGSS supports (but is not limited to) the current Public Key Infrastructure (PKI) deployed at the IAEA department of safeguards for user authorization. EURATOM has set up his own root certificate so that both parties can act independently, but are able to verify later the authenticity of all information or can see on the authenticated log file what was done and by whom.

## 3. NGSS System Design

For practical reasons the All-In-One-System (called internally XCAM now) was developed first (see Fig.2). It contains the SCC with the integrated CMOS sensor and the lens (called then DCM-C5), the backup battery compartment, the mains power supply and the human interface (5.5" colour monitor with jog-dial) in a special housing. The DCM-C5 is also able to work in a virtual 4 camera mode to replace up to 4 cameras on one place looking to different directions. All data leaving the SCC either via Ethernet or RS 485 connection or stored on the exchangeable SD card buffer is authenticated and encrypted using state of the art cipher suites.

### 3.1. Proposal from the Manufactures

The NGSS system consists of 1 to 32 XCAM connected to a DC (Data Consolidator – housed in a 19" cupboard, see Fig.3) [6]. Each individual XCAM corresponds to one DCI (Digital Camera Interface) located on the DC and equipped with its own UPS (Uninterrupted Power Supply). The DCI provides the DC power to the camera and pulls the data from the camera and sends it over the DC Ethernet network to the 2 (or more) removable NAS (Network Attached Storage). A CPU (Central Processing Unit) monitors the traffic, handles cross triggering signals and supervises (if installed) the RDT via a

VPN (Virtual Private Network) box or other modems. The NGSS GUI (Graphical User Interface) allows interfacing, set-up and maintenance. The hot swappable NAS drives can be removed from the DC and transported to a secured environment with a review station (standard PC with a NAS reader and GARS review software) for the image and data review. Preventing loss of surveillance has been the main concern for the design of the NGSS. As a result the NGSS records data in 3 places: on the SD card located in the XCAM, on the SD card located in the DCI, and on the NAS.


Figure 2: XCAM with open housing doors (front door for lens service, upper door with human interface and access to the SCC below, rear door with battery compartment)

After delivery of the 48 XCAM and two DC prototypes as part of the SP development and the successful test of these components, the purchase of different systems to replace the DCM-14 systems was planned. Due to the custom design development, the high performance of the XCAM and the high redundancy in the DC the proposed prices were very high. In the first estimate the price for a XCAM was about 24 000 € and for a 12 channel DC in the range of 120 000 €. That means that the average "channel cost" for a NGSS system would be about 34 000 €.

Due to the general economic situation in all public sectors the price efficiency in safeguards has become more and more important. With a frozen or reduced budget the IAEA and EURATOM are unable to spend more money for surveillance systems as done in the past. Based on this fact EURATOM estimated [7], that the average channel costs maybe not higher than about 12 000 €, otherwise the new system cannot be financed. The calculation was very simple: EURATOM cannot spent more than one million € per year for surveillance, has to replace in the next 10 years about 350 cameras and related systems in the EU-NNWS and about 400 cameras in the UK and France. Thus about 4,5 M€ are available for the replacement of the 350 DCM-14 cameras and system components. Even if the IAEA received in the past sometimes additional funds from certain countries the IAEA might also have problems to finance the proposed prices from their regular budget. That means the costs have to be reduced drastically. How can this be done? A reduction of the price by 60% can normally not be achieved by negotiation.

Figure 3: A 12 channel NGSS data consolidator with 3 NAS modules (middle right)

### 3.2. Improvements by the Users

Thus a redesign or reconfiguration of the NGSS by "downsizing" was the only way forward. The NGSS was investigated concerning all components and redundancy which is not essential for the proper function and reliability of the system with the goal either to get rid of them or maybe to replace them by commercially available components. Therefore the different components and their real importance in the future safeguard systems were evaluated. This work was done by the IAEA and EURATOM mainly independently, but the results are nearly the same.

If the XCAM will be used for All-In-One applications, i.e. for temporary surveillance or on places were only one camera is needed and the mounting place is easily accessible, all components of the XCAM are necessary. A price reduction is only possible by tough negotiation and high quantity ordering. And indeed both organisations could achieve a reduction of 5…35% for a full equipped XCAM, depending from the order quantity. Further reductions are possible for instance if the back-up battery will be directly purchased from the battery manufacturer, to avoid the dangerous goods handling and transportation requirement because the double Li-Ion battery pack has an energy content which is 2% over the dangerous goods threshold. Finally, the price for an XCAM is acceptable because it is in the same range (for a batch of 100 pieces) than the prices for a DCM-14 ALIS, but the performance of the XCAM is much better.

For single and multiple camera applications there is no need for the human interface and mains power supply in the camera module. This camera module is centrally DC powered and the set-up, control or

maintenance can be done from the central point on-site or remotely via VPN. Furthermore EURATOM decided also to reuse the existing DCM-14 housings because the NGSS housings are expensive due to the sophisticated rotating hinge with the electrical pass-through and lot of doors allowing easy maintenance access. Furthermore the cost expensive decontamination or decommissioning of slightly contaminated old housings by the operator can be avoided. The necessary camera parts will be pre-mounted on a component holder (including cable adapters), so that on-site activities can be reduced to the changing of 6 screws and reconnection of the existing cables (see 3.3.). The batteries will be purchased on demand directly from the battery manufacturer by a framework contract, so that they are always fresh and not degenerated by a long storage period. This component reduction and reuse, allows EURATOM to purchase a camera module (in a batch of 100 pieces) with the full needed functionality for less than 10 000 € where the battery price is already included. A further reduction might be possible by using the virtual 4 camera mode of the SCC, so that the total amount of cameras can be reduced. But this reduction might be only in the order of 10% and will be fully compensated, if for full redundancy at least 2 cameras per MBA would be installed.

The main reduction in price without losing essential functionalities can be made on the DC (data consolidator), because it is not a crucial component. The DC is foreseen mainly for data back-up and easy camera and data access as well as for concentration of data for RDT activities. The UPS strategy where each component has its own UPS makes the DC expensive and heavy, because left half of a 19" cabinet is full with heavy UPS modules. The power availability is crucial, of course, but the camera has already its generous designed back-up battery and the powering of DC components is not essential. This additional redundancy has the disadvantage that due to the weight distribution inside the DC cabinet the seismic qualification could become a problem. Furthermore with the availability of 128 GB SD cards in the DCI the additional storage of information on an exchangeable half TB NAS becomes questionable.

Figure 4: DCR-1 baseplate with power supplies (top left), DCR and DCI (middle right) and VPN box preparation (left) in a DSOS housing

The DCI was originally able to act as a client for up to 4 (virtual or real) DCM-C5 or up to 32 DCM-14 cameras and as a server for the CPU or RDT clients. Because more than 90 % of our multiple camera systems in the EU-NNWS have 8 cameras or less the idea was born to upgrade the DCI firmware to address and concentrate the data from at least 8 camera modules. The upgraded DCI together with a small human interface allowing simple set-up and SD card exchange and an industrial 24V power supply represents a cost effective solution that can be installed reusing the DSOS housings and so replacing the DSOS and most of the SDIS systems. The result was the DCR-1 (Digital Camera Recorder for 1 DCI). All components are pre-mounted on a new baseplate fitting in the DSOS housing and the on-site activities for upgrades are reduced to the baseplate exchange. As the housing is reused this approach reduced significantly the upgrade effort in several installation. For more sophisticated set-up or maintenance, a PC with dedicated software can be connected to the DCI upload Ethernet connector to interact with the DCI and all connected cameras.

The costs for such a data recorder or concentrator (either the full equipped baseplate or for components to be mounted on DIN-rails in the existing SDIS enclosures) are less than 10 000 €. This means that with an average of about 4 cameras (single or multiple camera systems) the average channel cost is coming down to the targeted range of 12…13 000 €.

For larger systems either several DCI can be mounted, controlled from an industrial PC with high storage capacity or - since the firmware development of the DCI is not completely finished – another smart solution could be implemented. In this case the channel cost could be further reduced; a DCI will be shared by 6…8 channels and the costs for the industrial PC, the power supplies and 19" rack holder are in the range of the cost for the DCR-1 baseplate without the DCI.With this redesign the costs are reduced to the available financing level. Therefore EURATOM has concluded a 4 years Framework Contract with Dr. Neumann elektronik GmbH enabling the purchase of 300 cameras and all the necessary components for the replacement of the existing systems in the EU-NNWS in the next 5…8 years. The IAEA has purchased already more several hundred cameras and components and will soon conclude also a Basic Supply Agreement.

### 3.3. Implementation Details

One of the most under-estimated costs is the cost for new cabling in a nuclear facility. These costs including project planning, documentation, licensing and supervision of external service providers often exceed the cost for a whole new multiple camera surveillance system. Therefore the reuse of existing cabling was one of the essential design criteria for NGSS so that normally all replacement work can be done by EURATOM or IAEA technicians.

The existing different cable types include many variants of twisted pair-, RG-58 or RG59 coax- or modern cat.3…7 Ethernet cables. Due to the built-in TCP/IP protocol via Ethernet or high speed RS485 for data transmission from DCM-C5 to DCI all existing cables can be used either directly or with industrial available cable converters, i.e.

a) Ethernet over Cat.3...7 cable (2x Unshielded Twisted Pair) up to 100 m distance – to avoid crimping in field, an E2TP (Ethernet to Twisted Pair) adapter might be used,
b) Ethernet over RG58- or RG59-Coax cable up to 500 m distance using E2CC (Ethernet to Coax Converter) on both ends,
c) Ethernet over Fibre optic (62.5/125 µm) cable using E2FO (Ethernet to Fibre Optic) on both ends,
d) TCP/IP over RS485 cable (1x UTP) up to 1200 m using existing Phoenix connectors on DCM-C5 and DCI where the bus structure can respected, otherwise a special RS485 adapter (DCS from manufacture, based on DCI hardware) must be used before the DCI.

For the 24 VDC power transfer a separate cable pair with 0.75...2.5 $mm^2$ is mostly available. For special cases the power transfer over remaining UTP lines of an Ethernet cable (also known as PoE) might be used, if it was previously tested with the highest load (full recharge of empty batteries).

Following the system optimisation the main issue for the NGSS implementation in the future will not be the financial issues anymore; it will be the human resources to perform the replacement. The implementation of the DCM-14 technology lasted several years, it is finished now and it was a big

investment of financial and human resources. For sustainability reasons existing DCM-14 camera should be used until the end of their life time if the improved features of the NGSS are not requiring an earlier exchange. Sufficient DCM-14 spare parts are available since the failure rate of the DCM-14 is less than 2% in 5 years, much smaller than expected.

The excellent reliability and the investment in the DCM-14 based technology are motivating EURATOM to perform the necessary preventive maintenance activities after 4…5 years' operating time and to upgrade the DCM14 based SDIS multiple camera systems by adding an additional 24 VDC power supply (PS) which is connected directly to the operator mains. This will avoid the loss of surveillance after the backup batteries are exhausted in case of failure of the original 24 VDC PS or of the UPS (Uninterrupted Power Supply). After this upgrade the SDIS will be protected against single point of failures. The SDIS upgrade will also include some software to enhance SDIS user-friendliness. The APCUPSD software records the state of health parameters of the UPS system allowing the technicians to get information about its status and, if RDT (Remote Data Transfer) is implemented, to reset and control the entire system. An easy to use inspector interface allows the inspectors to set-up or check the system locally, i.e. to set-up other PTIs (Picture Taking Intervals), download the latest images to the EHD (Exchangeable Hard Disk), perform on-site reviews of images, or transfer image files manually from the internal HD to the EHDs.

The limited RDT capabilities of the DCM-14 based DSOS system is in the moment the most important driving force for the upgrade to the next generation. After solving the SEU issue in 2000 the DCM-14 firmware development was frozen after intensive tests without implementing all features for the RDT. Few years ago, as RDT features became important, it was recognised that the firmware development tools are not available anymore. Fortunately due to the required backward compatibility the DCI can also collect data from 1 to 32 DCM-14 connected in serial bus structure or where star topology exists with the help of one or 2 RS485-4 hub splitters. If a replacement of a difficult to access DCM-14 camera will cost too much money and/or time due to expensive scaffolding works, even mixed systems with several DCM-C5 and DCM-14 cameras are possible by using 2 DCIs. Thus a smooth migration to the NGSS can be adapted to the available financial and human resources.

There are 2 ways to make DSOS RDT capable. Where 2 or more DSOS exist per MBA, both cameras can be connected to one of the DSOS housing and the other can be removed. On the remaining DSOS housing the original baseplate will be replaced by a DCR-1 baseplate (see above and figure 4). If several MBA exist on one site, with only one DSOS per MBA it is also possible to keep the DSOS nearly unchanged by adding only in the left upper corner a timer module. This module is switching during its passive phase the RS485 bus from the DCM14 camera always to an on-site centrally installed DCR-1. So all cameras are remotely accessible via the VPN and DCR-1 and the DCR is collecting and buffering all images. Twice a day or if an inspector wants to service the DSOS, the timer can be manually activated and is then connecting the camera RS-485 bus to the DSOS This allows all images to be stored in the DSOS locally. This configuration was tested successfully last year in the EURATOM HQ. This method saves human effort and DCR-1 units on sites with a lot of MBAs and is useful especially in the beginning of the replacement campaign.

## 4. Current Situation

As part of the development, the NGSS system was comprehensively tested. The Vulnerability Assessment was performed by the Joint Research Centre of the European Commission, the environmental tests were performed by the manufactures and the irradiation tests were mainly done by the IAEA with support from EURATOM. All these tests provided successful results. The practical behaviour of the firmware was tested in the headquarters in Vienna and Luxembourg.

The development of the SCC firmware is finished now, the final tests of FW 2.11 were successful and the firmware is frozen and authorised for inspection use. The development of the DCI firmware is nearly finished. The last tests on FW version 3.00 are on-going. Nevertheless the firmware will be further developed to ensure its compatibility for large multiple camera applications. These tests have and will be done mainly independently by IAEA and EURATOM but in case of detected failures the other party will be informed immediately.

The IAEA has already a lot of lab and in-field experience with XCAM, XCAM-DCR-1 and DCM-14+DCR-1 based applications because they could use the 48 XCAM, DCR-1 and the DC prototypes

as part of the development phase since 2010. EURATOM has borrowed 2 XCAMs from the IAEA and has purchased few a XCAM and DCR-1 modules for test purposes in 2012. Some common workshops have been set-up to learn from each other and to coordinate the work and planning together. Examples of this cooperation are the jointly performed camera keying exercise to test the keying procedure, several workshops to get practical experience in setting up the camera and working with the cryptographic certificates. Also the SD card model to be purchased and the test of the SD cards have been agreed because the reliability and proper working of the SD card is crucial for the data storage on the DCM-C5.

For the common use of the NGSS a Partnership Approach document "Joint Use Arrangement for NGSS" (JUA) has been drafted. Here all relevant activities and practical arrangements from the camera keying, set-up, installation, transport and troubleshooting up to the working papers and the review activities are described. A simulated joint inspection was organised to gain more experience on use of the system. This information was used to improve the JUA document. Finally a common field test in WKKG is currently performed to prove in practice the joint activities including the safeguards review.

The implementation of new technology requires always a dedicated training of the users. The first inspector training was performed to make the inspectors familiar with the data security requirements, the set-up of certain camera parameters and the SD card exchange. The first feedback was quite positive because the inspectors find the NGSS menus intuitive and sufficient warnings or explanations are popping up, if a risk of mistake exists. The Working Paper will be updated soon to make it better understandable for inspector dealing the first time with the NGSS components. Further trainings are already planned to train more of the inspectors. However, also the technicians need further dedicated training especially on the keying and certificate handling. Therefore an advanced training where the IAEA technicians are invited is planned for June to spread the available knowledge to all technicians.

## 5. Further Steps

At the moment EURATOM and IAEA are establishing the replacement plans. It has been agreed that EURATOM takes care of all nuclear installation in the EU-NNWS except of the on-load reactors. The IAEA will replace and equip the installation in the rest of the world and the on-load reactors in EU-NNWS, because they need a permanent surveillance there.

The replacement will start with ageing DCM-14 systems, which are malfunctioning, need comprehensive maintenance or have limited capabilities. This concerns mainly the ageing ALIS or ALIP (an ALIS with additional batteries for operation without mains for temporary surveillance) which will be replace by XCAM. Where RDT capability is needed and a DSOS is installed the DSOS will be either upgraded like mentioned in chapter 2.3 or the recording unit will be replace by a DCR-1, depending from the number of cameras and age of the system. If the DCM-14 cameras have nearly reached the end of life time they will also be replaced by DCM-C5 modules, forming a complete new NGSS system (called then XSOS). Recovered DCM-14s which are exchanged for practical reasons, but not used for at least 7 years, will be used as spare parts for the remaining systems to cover sudden exchange requests.

In the next step the SDIS systems will be upgraded or exchanged. The PIP computer, the power supply and the UPS will be replaced by DCR-1 components. Ageing IAEA system will be completely exchanged (called then XDIS). EURATOM SDIS systems are only few years in field and will be first upgraded as long spare parts are available. So the SDIS replacement by NGSS will start in 2015 and last until 2020.

For the few DMOS systems in EU-NNWS with more than 8 connected cameras special solutions have to be investigated. Either the firmware of the DCI can be improved to handle more than 8 cameras or several DCI will be used and connected to an industrial PC with sufficient mass storage capacity and software for set-up, interacting and troubleshooting the connected DCI and cameras as well as for on-site image review. The PC with display, keyboard and mouse is either installed in a 19" cupboard in a convenient place or in larger installation with an on-site inspector office maybe also as a desktop version in the inspector office (will be called XMOS system - see figure 5).

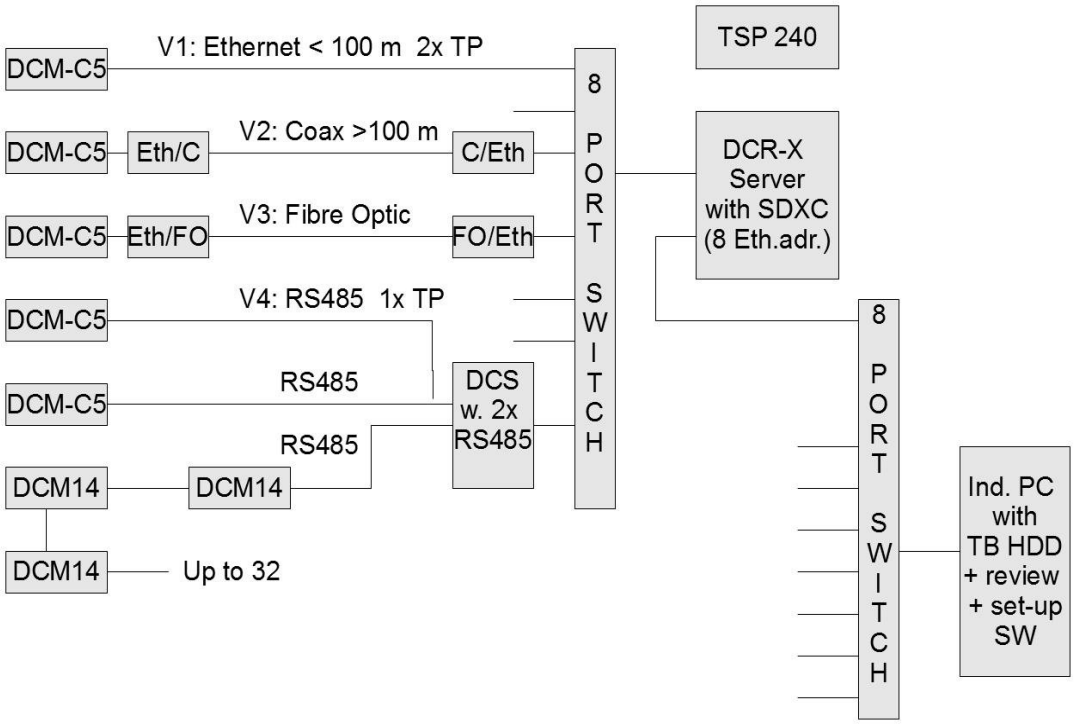


Figure 5: Possible XMOS system with demonstration of reuse of different existing cables

For underwater application a prototype of an underwater housing with build in DCM-C5 and battery pack is currently tested by the IAEA. The camera has a remote zoom, its own light source and can be adjusted from above water level in 2 axes, so that all possible angles of view can be achieved.

One of the outstanding issues is the connection of analogue CCTV cameras the NGSS systems. This requirement comes from special cases where radiation hardened or very small cameras have to be connected to the surveillance system. In some cases also a DCM-14 can be used but there is the general problem that the cable from the camera to the SCC is vulnerable for image falsifying. First ideas exist to check this cable electronically from the SCC against manipulation. This development will be done under a Ge-SP task for field support for the NGSS system implementation.

## 6. Summary

The DCM-14 based systems are a very reliable surveillance tool, been in field since 2000, fulfilling all requirements for safeguards equipment and it will be further used in the next years. But due to the non-availability of the IC generation and software development tools as well as some limitations for RDT applications the next generation of surveillance systems had to be developed. The development of the hardware components and the firmware for vital parts, supported by the German and US support Program for the IAEA is now finished and the components have been successfully tested. The result is a NGSS system with the same modularity, flexibility and reliability as of the DCM-14 system, but with improved features for image taking, power autonomy, RDT capability and mainly for data security and tamper indication.

During the purchase phase it was recognised that the original proposal of the developers cannot be financed with a zero growing or reduced budgets. Therefore a redesign was suggested by EURATOM and the IAEA to reduce unnecessary redundancy while keeping the full functionality. As a result the price of the camera systems went down up to 40% of the original price. The purchase and production is on-going now, the IAEA has received already a big amount of NGSS components, EURATOM is awaiting them in the second half of 2013.

Due to the backward compatibility of the NGSS components not all components have to be exchanged at once. So the DCM-14 cameras can remain in place until the end of its lifetime which makes the full use of this investment possible and allows adapting the replacement speed to the human and financial resources available. That will also allow the inspectors to get familiar with the new system step by step, because not all of the staff can be trained at the same time.

First the ageing, maintenance requiring or malfunctioning equipment will be replaced. This concerns mainly the ageing ALIS or ALIP (an ALIS with additional batteries for operation without mains for temporary surveillance) which will be replace by a XCAM. Where RDT capability is needed and a DSOS is installed the DSOS will be either completely upgraded or only the recording unit will be replace by a DCR-1. The SDIS systems will be upgraded before they will be replaced by NGSS.

Due to the very good cooperation and information exchange between the technical surveillance teams from IAEA and EURATOM it is possible to learn from each other, share the work load and to avoid mistakes during the implementation. These good relations have significantly helped to come up with an affordable yet modern surveillance system. These relations must be maintained to allow the big replacement work load to be performed with the limited human resources within the next 5…8 years.

## 7. Acknowledgements

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# Free form source representation for a VR dosimetry training application

**Teófilo Moltó Caracena[1], João G.M. Gonçalves[1], Paolo Peerani[1], Eduardo Vendrell Vidal[2]**

1. Joint Research Centre, European Commission
2. Universidad Politécnica de Valencia
E-mail: teofilo.molto-caracena@jrc.ec.europa.eu, joao.goncalves@jrc.ec.europa.eu, paolo.peerani@jrc.ec.europa.eu, even@upv.es

***Abstract:***

*A method to model free form nuclear sources in Virtual Reality (VR) based dosimetry applications for training purposes is presented in this paper. A VR based dosimetry application can provide advantages in terms of time, cost and logistics with respect to traditional on-site training courses, representing a valid complement to the traditional ones.*
*In order to benefit from these advantages, methods need to be developed to overcome the requirements such a VR application requires, such as real time response and accuracy of dose rate computation. The developed method based on the prototype [1] provides both but limits the kind of source geometries that can be simulated, only allowing parallelepiped source volumes.*
*This paper presents a new version of this prototype which implements a revised algorithm which can cope with free form sources, allowing the tool to be able to cope with other kind of geometries which are bound to be found in a real case like cylinders or spheres.*

**Keywords:** Virtual Reality; Training; Nuclear Safeguards; Nuclear Security; Computer simulation

## 1. Introduction

During the last twenty years VR methods have been used extensively and successfully in many fields science and industry, including nuclear research and science [1] [2].

Particularly, VR has been used for training purposes in this field [4] because VR based applications can provide advantages in terms of time, cost and logistics with respect to traditional on-site training courses, representing a valid complement to the these.

In the case of this project the aim is to train subjects with scarce knowledge in radiation physics in the use of radiation detection equipment. This might be necessary for professional groups such as first responders to accidents (e.g. firemen, police, civil protection etc.) or customs officials in charge of border security.

The work presented on this paper is a continuation of the prototype presented in [3], which showed a VR based simulator of survey meter (a handheld portable dosimeter) for training purposes. This is the latest enhancement of a prototype which first started in 2009 and has been continually developed since.

Despite the prototype met the specified targets, some limitations were found associated with the geometric shapes of the radiation sources. In this application solutions to this source shape limitation are explored.

The proposed solution is based on a mixed strategy which combines the original source volume division method created (irregular Solid Angle driven) which targets computation efficiency with a well-known method in computer science visualization of volumes: Boundary Representation (BREPS) which aims at creating an accurate source shape representation.

## 2. Solid Angle driven volume division

The latest version of the prototype featured a dynamic, real-time algorithm to automatically divide the volume of the radiation source in a computationally efficient way. This contrasts

with the usual regular division most dosimetry software programs use.

This novel method exploits the fact that radiation decreases exponentially with distance. Therefore an irregular division of the source can be created by increasing the resolution of the division in more significant areas and keeping the current coarse resolution in the rest of the volume. The following figure shows an example of the same volume, in one case divided by a standard regular division method and on the other by the novel method.

It can be noted that the novel method generates a representation with many less sub-sources than the regular division. Considering the speed of the computation is directly proportional to the number of sub sources present, the new method represents a significant improvement in computational cost.

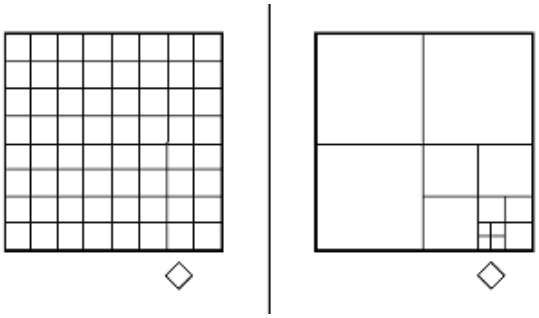


**Figure 1** regular vs. solid angle driven division (rhombus represents position of detector).

For this method to be valid it must recognise which areas of the volume are less relevant to the dose rate computation. This is done by calculating the solid angle of each source (and their sub-sources).

The solid angle is defined in for our purposes as the area occupied by the source with respect the whole field of view from the detector's point of view. This parameter is chosen as it can account for variations in three significant (for the dose rate value) parameters: size, distance and orientation.

The following figure shows how variations in each of these three individual parameters induce a variation in the solid angle of that source. There is an exception when dealing with extremely elongated sources which do not represent a realistic case.

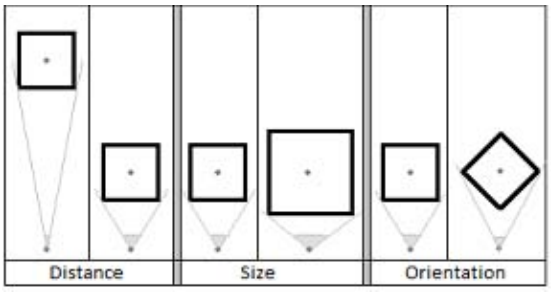
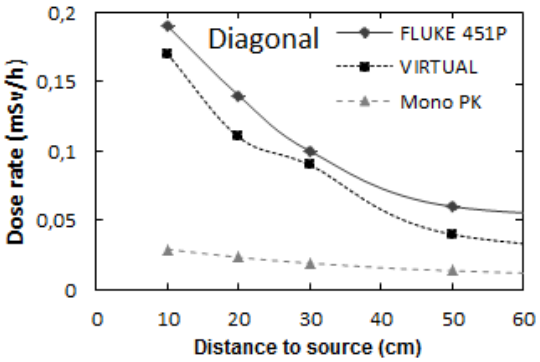



**Figure 2** Solid Angle variations induced by changes in distance, size or orientation.

When tested against real detectors and other computer simulation codes this method yielded satisfactory results in line with the requirements set in terms of deviation of the dose rate measurement.




**Graph 1** VR Solid Angle method vs. Real detector (Fluke) and vs. Static computer code (Mono PK).

Nevertheless a series of limitations were noted. One of them is the fact that only the radiation sources of parallelepiped shape have the boundaries correctly represented. Other common shapes for radiation sources, such as spheres or cylinders, would be represented as parallelepipeds. This simplification can lead to significant errors when taking measurements at close distances due to the poor representation of the boundary of the source. Next figure shows how parts of void are treated as part of the source volume for a cylindrical radiation source representation, incurring in a probable deviation if the detector is placed near or within this volume of "fake" source.

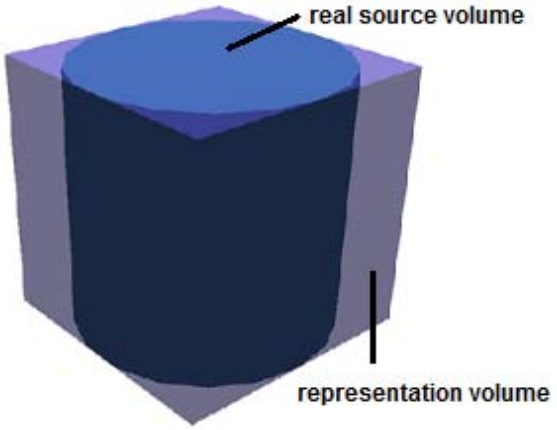



**Figure 3** The representation of a cylindrical source

A more accurate representation of the boundaries might be necessary to meet the deviation requirements of the application when treating non parallelepiped source shapes.

## 3. BREPS

The idea behind Boundary Representation (BREP) is to represent volumes of solids by their physical limits [5]. In the particular case treated in this paper the aim is to represent sources of multiple forms accurately.

BREPs represent the solid by a finite element cellular decomposition of its boundary. This provides a very flexible method to represent all sorts of shapes, the accuracy of the representation can be increased by simply increasing the resolution of the division. The following figure shows a coarse resolution BREP of the cylinder from figure 3.

The downside of this method is that it applies the algorithm on the whole volume of the radiation source, without considering the radiological importance of the different areas of the source. This might increase the computational burden on the system, which is not a wanted characteristic in VR based applications.

Our main purpose is to design an accurate and fast enough radiation simulation tool, i.e., capable of providing the sense of real-time visualisation and measurement feedback. As such, it would be convenient to control where the BREP acts in order to limit the associated computational costs.

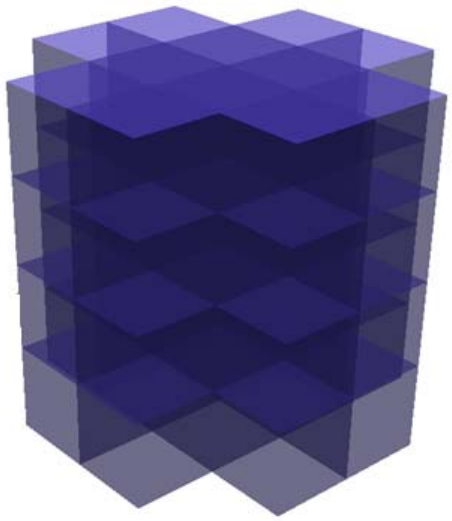


**Figure 4** BREP representation of a cylindrical source (48 sub sources)

## 4. Mixed BREPs + Solid Angle driven volume division.

Therefore the proposed solution is to combine both Solid Angle driven volume division with standard BREP boundary representation.

This is done by first applying the Solid Angle division algorithm and in a second step applying the BREP to the result of the Solid Angle division representation. This way the computational burden is reduced because the BREP works on a smaller sub group of elements.

The following figure shows and example of this mixed Solid Angle plus BREP strategy for the same cylindrical radiation source of Fig 4. It can be noticed the number of sub-sources to be processed by the dose rate computation algorithm have been reduced. This is because the algorithm takes into account the position of the detector. This allows the system to use a coarser resolution on the back of the volume, the sections which are further away from the detector (and therefore have a bigger solid angle).

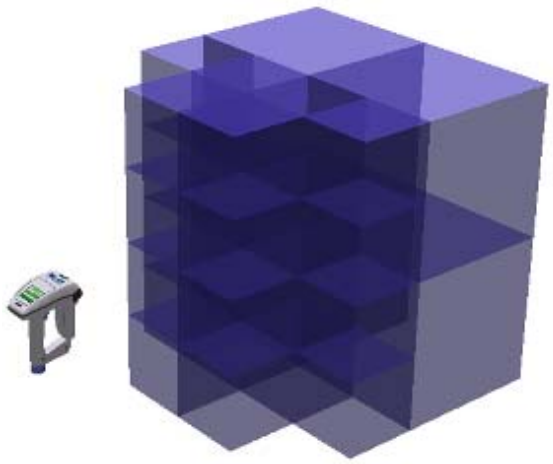


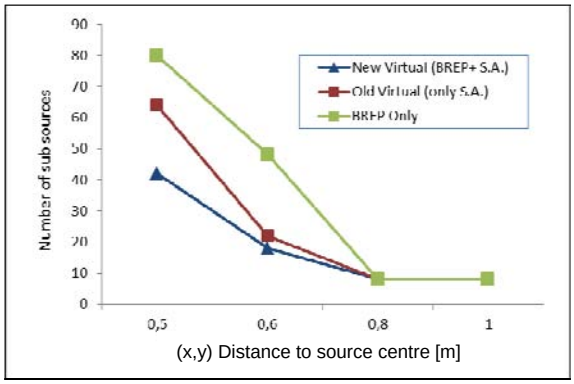
**Figure 5** Solid Angle driven BREP (28 sub sources)

## 5. Results

### 5.1 Computational cost test

The method was tested to evaluate the performance gained by reducing number of sub source elements.

A simulation was performed where the detector was approached on a diagonal line to the cylindrical source. Several measurements were taken varying the distance. The following figure shows the results obtained by the new combined method with respect to the previous one and a standard BREPs representation.




**Graph 2** New BREP + Solid angle model size vs. Old only Solid Angle vs. BREP only.

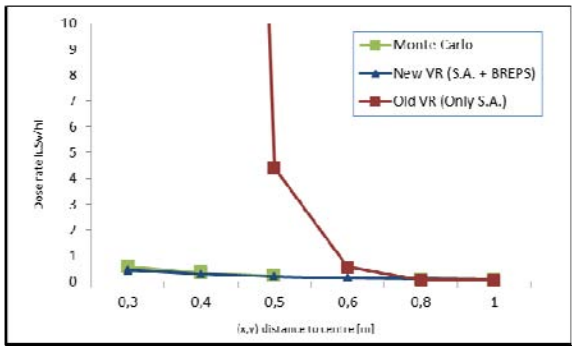
### 5.2 Dose rate accuracy test

The method was also tested to evaluate if the expected accuracy gain was real.

Similarly to the computer test, a series of measurement positions where established on a diagonal line approaching to the source centre.

A Monte Carlo simulation was performed to use as a benchmark which to compare the new (Solid Angle + BREP) and old VR (only Solid Angle) models.

The following graph shows the computed dose rate values at the measured distances.




**Graph 3** New (BREP + Solid angle) model accuracy vs. Old (only Solid Angle).

## 6. Conclusion and Future work

A noticeable improvement was noticed in terms of reducing computational cost, concretely at close distances. The number of sub sources generated by the new model both respect to the old only solid angle division (up to 35%) and a standard full BREP representation ( up to 48%) for the cases tested.

At long distances the results show no improvement due to the fact the model is the same because the solid angle of the source doesn't reach the threshold, but at close distances the improvement is notable as expected.

In terms of dose rate accuracy a massive improvement was noted in the expected areas shown in figure 3. Therefore successfully eliminating the limitation of only parallelepiped shape sources of the previous prototype.

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# Free form source representation

## for a VR dosimetry training application

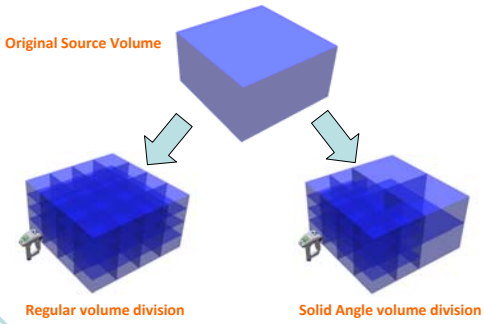
**Teófilo Moltó Caracena**[1], **João G. M. Gonçalves**[1], **Paolo Peerani**[1], **Eduardo Vendrell Vidal**[2]
1 European Commission, Joint Research Centre, Institute for the Protection and Security of the Citizen, Nuclear Security Unit, Ispra, Italy
2 Universidad Politécnica de Valencia, Valencia, Spain

### Virtual Reality and Nuclear Security training

- **Detecting illicit trafficking of radioactive substances** is a task for which **Customs officers and emergency First responders** need to be trained for.
- Traditional methods for radiation detection training use real **radioactive sources.** This is both **expensive** and **difficult to manage**
- **Virtual Reality (VR) based training** complements traditional methods and has the advantage of reducing the need for the use of real radioactive sources. This brings benefits in terms of time, cost and logistics.
- On the down side, the **real time** requirement inherent to VR, needs of **efficient algorithms** for dose rate computation.

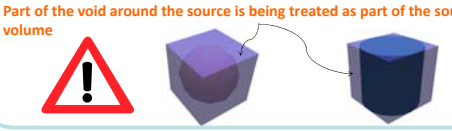
### Efficient Radiation Source Representation methods

To meet real time requirements a method based on **Solid Angle driven volume division** was developed, this reduced the number of points which represent a source in a Point Kernel dose rate computation with respect to the usual regular division of the source volume.



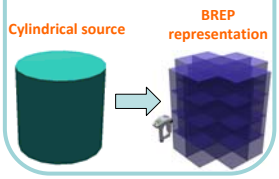

#### Method's Limitation

This method works with sources whose shape is a **parallelepiped**. But the method incurs in significant deviation if other common source shapes are considered such as **spheres or cylinders**.



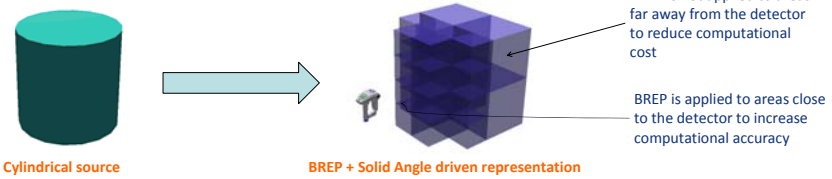

### BREPs

Boundary representation (**BREP**) techniques solve this problem, they can provide an **accurate** representation of source volume borders independently of the shape but lead to **high computational costs**.




### Proposed Solution

A **mixed BREP + Solid Angle** driven source volume representation method aims to be both accurate and computationally treatable by applying BREP representation only in significant parts of the volume.

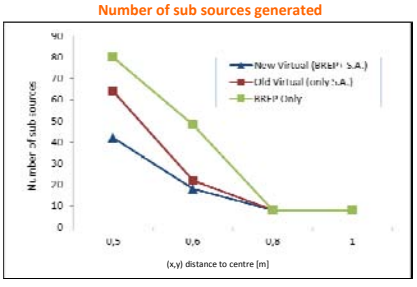



### Testing Results

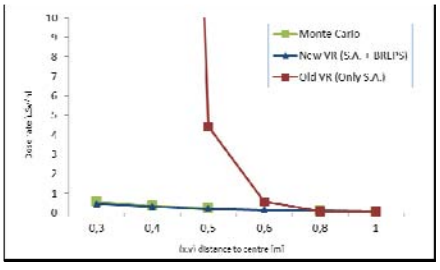
The new method was tested in the VR simulation and the results compared to the previous VR method to assess:

- Computational improvement (reduction in number of sub sources to compute)
- Accuracy of the Dose Rate computation, using as a benchmark a Monte Carlo simulation.

Number of sub sources generated




Deviation of dose rate computation




### Conclusion

- The new method **improves:**
  - **computational cost** (up to 50% reduction of sub-sources to process);
  - **dose rate accuracy** (within 20% deviation of Monte Carlo).
- Training application requirements for both accuracy and real-time computation are met
- This overcomes the limitation of the previous method.



**Contact:**
**Teófilo Moltó Caracena**
European Commission • Joint Research Centre
Universidad Politécnica de Valencia
E-mail: teomolca@inf.upv.es

**Contact:**
**João G. M. Gonçalves**
European Commission • Joint Research Centre
Institute for Transuranium Elements
E-mail: joao.goncalves@jrc.ec.europa.eu

JRC EUROPEAN COMMISSION

ITU Institute for Transuranium Elements

# *Impact of RDT on Future Safeguards Implementation and Approaches*

**S. Synetos, Y. Lahogue, C. Cosmo, C. Trichelot, A. Ozols, K. Schoop**
Directorate for Nuclear Safeguards, DG Energy, European Commission
L-2920 Luxembourg
E-mail: Sotiris.Synetos@ec.europa.eu

## Abstract:

*In recent years the efficiency of the EURATOM safeguards system has been drastically increased by re-locating a substantial number of the controls and checks from the sites to HQ in Luxembourg. This has become possible by making arrangements with operators and State Authorities for the transmission of operating records and of nuclear safeguards measurement or monitoring data to Luxembourg.*
*In the EU nuclear plants, especially the large bulk handling facilities, safeguards controls are based on a number of unattended systems, either purpose designed and built Euratom proprietary equipment, or the branching of operator's devices signals. Signals from unattended systems are normally transmitted to the inspector's office on site for further treatment, analysis and comparison with the operator's declarations, and could be selectively remotely transmitted to Headquarters for further use. Comparison at headquarters of remotely transmitted measurement or containment and surveillance data with operational and accountancy records requires that the latter are also transmitted remotely.*
*Remote Transmission of all Data relevant to a safeguards inspection is seen by Euratom as the prime way to improve the security of the transmitted data and facilitate their archiving and effective exploration at HQ. Meeting RDT requirements could also indirectly contribute to Operator's further stratification and consolidation of data provided. Remote Data Transmission (RDT) could also provide more flexibility for the operator when planning maintenance of equipment or other regulatory interventions in plant areas under C/S or safeguards monitoring.*
*This paper addresses the issue of the relation between RDT, Safeguards Activities at Headquarters and possible reduction of Inspection Mission Days (IMDs). The position taken is that RDT could be used so that inspectors (and inspection equipment) are better prepared at the onset of an inspection, allowing them to perform less repetitive tasks during onsite inspections while concentrating on BTC or supporting documents verifications or audit of operator's systems.*
*Inclusion of RDT in the safeguards approach of a plant must also consider the reliability of the remote connection, the effect of possible failures and the development of alternative approaches should a failure occur.*
**Keywords:** Remote Data Transmission, Safeguards approach, Headquarters activities

## Introduction

In earlier papers [1,2] the development and use of Remote Data Transmission for safeguarding bulk handling facilities and Pu stores in the EU was described.

Implementation of Euratom safeguards in general requires substantial resources. Recent evolutions of safeguards in the EU, especially the enlargement to 27 member states, the full implementation of the Additional Protocol and Integrated Safeguards by the IAEA, but also adaptation of the EU nuclear industry to the post Chernobyl and recently post Fukushima environment, have not rendered this effort less intensive, despite technical developments introduced by the inspectorates.

Changes in the Euratom safeguards approaches as reflected in the document Implementing Euratom Treaty Safeguards (IETS) but also operational adjustments have aimed at making the Euratom Safeguards system more efficient and effective. Among the measures taken, the use of unannounced or short notice inspections, combined with audits of the operator's NMAC system were proposed as the most promising towards an increased efficiency of the system.

On the other hand, technological developments, especially those related to unattended measurement and containment systems, have contributed the most in efficiency improvements. Use of remote data

transmission was also proposed, a number of developments occurred, but its full potential has not yet unfolded.

This paper will attempt to describe the experience of Euratom with RDT especially in large bulk handling facilities and the impact it could have on future safeguards implementation, including in safeguards approaches.

## Current use of RDT in bulk handling facilities

So far, Euratom has routinely used Remote Data Transmission in the safeguards implementation at bulk handling facilities and in particular reprocessing plants.
Reprocessing plants are characterised by their complexity and difficult access in many areas due to high radiation levels, by their continuous mode of operation requiring a high degree of automation, and the presence of significant amounts of direct use material both as throughput and inventory. Security considerations play also a significant role in limiting direct inspector presence for verification purposes.

As a consequence of the above characteristics of reprocessing plants, safeguards approaches based on the installation of unattended equipment operating in continuous mode, maximum use of containment and surveillance systems and most important, transfer of the signals/information collected either to a dedicated technical room or preferably to the Inspectors' office on site were conceived already in the 90s and 2000s [3, 4].

The existence of such central data collection configurations was further exploited at a more recent stage, with the remote transmission of some of the data collected towards the Euratom Headquarters in Luxembourg, and was complemented by the electronic transfer of some of the operators operating data to Luxembourg, making thus the comparison between operating records, inspectors' measurement observations and results, feasible at HQ.

The safeguards approach in a reprocessing plant and associated Pu stores is based on the verification of the flux of nuclear material, the annual Physical Inventory Verification, the evaluation of the MUF at the end of the material Balance period and the evaluation of the C/S measures for Continuity of Knowledge purposes.
Depending on the MBA type and the state of the Nuclear Material a number of verification activities are performed in order to implement the safeguards approaches, including:

- Item Identification
- Weighing
- Volume and density measurements and weighing of tanks
- Continuous monitoring of the liquid level in the tanks
- DA sample taking and analysis (concentration, assay, isotopic composition)
- NDA measurements (neutron and gamma)
- Neutron Portal monitoring
- Video surveillance and electronic sealing
- Use of Copper Brass metal seals

With the exception of the metal seal verification and the DA sampling, all other verification activities are based on continuously monitoring systems connected to a local acquisition system. Raw data are transferred to the inspectors' office on site, are analysed and compiled to the 'events database'. The latter is then compared with the operating records as declared by the operator.

Bigger nuclear plants have developed a site wide system of operating records, which apart from serving operational needs, also constitutes the basis of the operating records submitted to Euratom in fulfilment of the operator's obligations under Regulation 302/2005.
Considering the big number of 'events' that have to be reported to the regulator, operating records are often produced and transferred electronically.

Implementation of RDT consists of the transfer to the Euratom Headquarters in Luxembourg of both the 'events database' and the operator's operating records and provisional reports. The former is done using a remote connection, while operating records are copied to a secured USB stick and transferred to Luxembourg by the inspectors.
The possibility thus exists to complete the inspection activities in Luxembourg.

Current Euratom inspector presence in Reprocessing Plants (number of inspectors and frequency of inspections) is rather related to the bulk of the information provided, the time needed to process it and the requirements of the operator to access part of the installation for maintenance of equipment than any nuclear material timeliness criteria.

## Issues and solutions around RDT

Operational and accountancy related data referred to above are considered as sensitive and carry a confidentiality classification at a level that might vary from one member state to another. It is thus important that for each category of data, Euratom, the State Authorities and the Operator involved, come to an agreement on exactly which data and in which way it will be transmitted so that the security requirements are met. Particular precautions have to be taken when data already classified are put or transferred together (e.g. data pertaining to nuclear material weights, location and time).

So far, for reprocessing plants in the EU, Euratom has obtained the agreement of the competent authorities for the remote transmission of some Euratom C/S, NDA and other equipment data from the UK (Sellafield) [5], while Operator's records and supporting documents are transferred using secure USB sticks from both Sellafield and La Hague. Discussions aiming at reaching an agreement on the scope and security requirements of RDT from French installations to Luxembourg are on-going. A security plan is in preparation and discussion, describing in detail the data to be transferred, the security requirements, vulnerability and threats, a risk matrix concerning the confidentiality, availability, authentication and integrity versus the proposed hardware, media, network and organisation structure. It turns out that the existing technology for secured data transfer allows nowadays a much safer transport of data via the internet compared to the transport via USB sticks or other media.

Details of the technical and security arrangements agreed have been presented on earlier occasions. In order to avoid issues related to the volume of the data transmitted and the synchronisation of the data bases in Luxembourg and on site, the preferred option would be to store all data on dedicated servers on site and access them via remote secure protocols from Luxembourg. The same servers would serve as the depository of the operator's data currently scanned and transferred via USB to Luxembourg.

The security arrangements at Headquarters and the easiness of access to the data by the inspectors are two competing requirements, which for the moment are addressed by restricting the availability of the remotely transmitted data to a dedicated server located in a room accessible only to inspectors on a need to use the data basis.

Higher dependence on RDT brings up the issue of reliability and performance (bandwidth) of the remote connection and the existence of appropriate back-up. Technical solutions based on simultaneous use of several telephone lines, or the use of satellite transmission are under development.

Once the above issues would be fully addressed, the advantages of RDT could be used to their full extend. Such advantages include:

- Increased data security through appropriate secure transmission
- Better possibilities of organising, exploring and archiving the data
- Data consolidation and stratification through early agreement on which data need to be provided
- Better possibilities to develop programs for data analysis and evaluation, based on the agreed format and context of data transmitted
- Integration of all information needs in one file, with the resulting ease of access and analysis
- Increased efficiency through harmonisation and streamlining between facilities of the same kind
- Improved conservation and transfer of inspection know-how and the cross-fertilisation between teams of inspectors

## Optimising the inspection regime

As already mentioned, reprocessing plants are characterised by high quantities of nuclear material throughput and inventory, and a big amount of data (operating records and measurement or C/S data) available to the Regulator often remotely.
Thus, the question of use of this plethora of data towards improving the efficiency and the effectiveness of safeguards verification activities becomes evident.

Recent developments of Euratom's inspection regime in bulk handling facilities point towards a combination of inspection verifications on site and inspection activities performed remotely at Headquarters.

Activities that could be performed at Headquarters to support or to prepare inspection activities include the following:

As far as equipment is concerned:

- Follow up of the state of health of equipment and planning of interventions, immediately after a failure is observed
- Planning of alternative safeguards measures in case of equipment failure or loss of containment (e.g. review of back-up video system)
- Remote update of system parameters (mainly software)
- Remote checking of completion of maintenance or upgrade actions requested by the inspectors (e.g. transducers calibration, or testing of proximity switches)

As a result of the above, the operator would have higher flexibility in planning and performing maintenance operations, and could have enhanced access for maintenance of his/her equipment installed in controlled areas (currently limited during inspector presence). Reduction of dose uptake for the inspectors but also for the operator via better planning is obvious.

As far as information is concerned:

- Review of surveillance records and reconciliation with events as declared by the operator
- Review of electronic seal events and reconciliation with events as declared by the operator
- Evaluation of continuous measurements (e.g. accountancy tank volume and density, or neutron and gamma measurements) and comparison with operating records.
- Evaluation of unattended NDA measurements of discrete items against the operator's declarations
- Comparison of operating records and reports with accountancy declarations received in Luxembourg

As a result of the above, possible discrepancies will be detected and analysed before the arrival of the inspector to the installation, giving adequate response time to the operator.

The question now arises whether the activities at Headquarters described above could replace the physical presence of inspectors at facilities, and could thus be considered as a safeguards inspection executed remotely.

The answer to this question could be different, depending on the type and characteristics of an installation:

For example, in static stores with effective C/S (e.g. a multilayer containment and ideally a robust, reliable and failsafe outer containment layer (door switch), a number of physical inspections could be substituted by Remote activities without either jeopardising the possibility of the operator to access the store for equipment maintenance, or increasing the probability of follow up measures due to C/S failures.

On the other hand, in bulk handling facilities, the physical presence of inspectors would have to be continued at a fairly high level, but the emphasis of their work on site would be shifted to activities like BTC re-verification, examination of supporting documents, audits of the operator's NMAC system, and

resolution of anomalies. It should also be noted that physical presence of inspectors familiarises them with the installation and the processes, and is of utmost importance when verifying the BTC or when introducing new inspectors to a new facility. Direct contact with the personnel of the installation should also not be underestimated as a positive factor for problem solving. Inspector presence serves also as a deterrent for potential diversion and promotes application of good operational practice.

Another factor to be considered is related to the optimisation of the inspector's time, both at Headquarters and during inspections. It can be argued that remote inspection activities could lead to a more efficient use and a homogenisation of the inspector's work pattern at Headquarters, and an increase of their efficiency during inspections (through better planning and reduction of unforeseen difficulties). Reduction of cost (less travel costs, but also less time lost for travel) is also a factor to be taken in consideration.

For the operators, remote inspection activities could only lead to advantages, related to the flexibility in planning their equipment maintenance interventions and the predictability of the inspectorates' requirements during inspections. Possible reduction of physical presence of inspectors would also reduce the need for the provision of escorts and operational staff for manipulating nuclear material, and would thus have an indirect positive effect.

An issue that will need to be addressed is the integration of the verification and comparison activities performed at Headquarters to the overall safeguards conclusions reached as a result of an inspection. A procedure replacing the current "opening meeting" i.e. the clarification of the scope of the verifications, the transmission of operating data and the discussion of any current issues potentially influencing the verifications, should be introduced. The format can however be more mechanistic and consist of a functional mailbox and standardised checklists per type of verification. The same applies for the 'final inspection meeting' during which the preliminary results of an inspection are presented to the operator. The latter will have to be substituted by a different means of communication (e.g. videoconference) which will allow the Inspectorate to inform the operator of the preliminary results of the integrated verification activities at Headquarters and on site.

## Remote safeguards approaches?

Notwithstanding further technical developments and finalisation of agreements with member states and operators, it would appear that all the necessary elements for the introduction of safeguards approaches based on RDT are in place.

Such an approach should describe the equipment necessary for its successful implementation, all data transfer requirements along with an inspection scheme and a matrix of actions in case of failure of some components of the system. All parties involved (i.e. operator, Euratom and the IAEA if it were the case) would then have a clear guidance of their rights and obligations under the approach.

As an example, an extract of a draft approach for a static NM store as developed by the IAEA and Euratom is presented in table 1: it shows that when the appropriate equipment is installed and RDT is applied, a big flexibility concerning the presence of inspectors can be introduced, making the safeguards system more efficient and effective.

Table 1: Draft Regulator action matrix for a static store under RDT

| **Remotely monitored** | | | **Evaluation based on remote verification** | **Access informed by EC/operator** | **EC Inspector On Site** | **Remarks** | **In-situ** |
|---|---|---|---|---|---|---|---|
| **EOSS on access doors** | **Surveillance** | **EOSS on channels** | | | | | **COBRA Channels (Static backup)** |
| ✓ | ✓ | ✓ | ✓✓ | - | - | | |
| ✓ | ✓ | ✘ | ✓✓ | - | Y | Request EC to investigate and perform correcting action, could access store if needed. | |
| | | | | | N | Request EC to investigate and arrange for corrective action. | |
| ✓ | ✘ | ✓ | ✓✓ | - | Y | Request EC to investigate and perform correcting action, could access store if needed. | |
| | | | | | N | Request EC to investigate and arrange for corrective action. | |
| ✓ | ✘ | ✘ | ✓✘ | - | Y | Request EC to investigate and perform correcting action, Supply EC with a list of COBRA seals to be verified (random low detection probability). | Y |
| | | | | | N | Request EC to investigate and arrange for corrective action. If Agency is not present at corrective action, EC to perform COBRA verification as above | Y |

The Remarks column shows immediate first reaction and is not meant to list all the actions that may be considered by the inspectors.

## Conclusions

Remote data transmission offers a number of advantages related not only to the better organisation and the security of data transmitted, but also to a better and timely preparation of inspections – reducing thus unforeseen problems.

Introduction of RDT as an element of safeguards approaches could be advantageous to all involved. Under such an approach, the range of inspector's activities at Headquarters and during inspections will have to be revisited so that an optimisation of the efficiency and the effectiveness of the system are achieved without any undue extra burden on the operator.

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# A Virtual Reality based Safeguards Surveillance Training Tool

**Teófilo Moltó Caracena[1], João G.M. Gonçalves[1], Susan E. Pickett[2]**

1. Joint Research Centre, European Commission
2. International Atomic Energy Agency
E-mail: teofilo.molto-caracena@jrc.ec.europa.eu, joao.goncalves@jrc.ec.europa.eu, S.Pickett@iaea.org

***Abstract:***

*This paper presents a Virtual Reality (VR) based application to provide awareness and basic training on surveillance concepts to future IAEA inspectors and analysts. This application benefits from the advantages of VR in terms of interactivity and variety of media to provide a complementary approach to classical training methods. This software application package includes a controlled area scenario and lets the user simulate and, ultimately, select the cameras, their position and orientation, their optical settings, and picture taking interval. One of the features is the possibility to simulate the motion of objects in the scenario, e.g., casks, and observe on whether daily object motion / transportation compromises the effectiveness of video surveillance. This application tool was recently used in a two pilot training courses at the International Atomic Energy Agency (IAEA). The scope was to raise and refresh the awareness of staff, inspectors and analysts– to basic Safeguards surveillance concepts and planning the installation and setting of surveillance equipment. Additionally, the application may be used as a daily tool for determining the best settings of the surveillance equipment. The paper describes the requirements at the origin of this application, and implementation details (including the software and hardware platforms that were used). It discusses key technical (i.e., software, hardware) requirements for virtual environment training tools for the department of safeguards based on the development process and user feedback from the pilot courses held at the IAEA.*

**Keywords:** Virtual Reality; Training; Nuclear Safeguards; Surveillance; Computer simulation

## 1. Introduction

During the last twenty years VR methods have been used extensively and successfully in many fields of nuclear research and science [1] [2]. These include Nuclear Safeguards and Security [3], where there are foreseen advantages in terms of efficiency, cost reduction, training time and safety. Further, it is expected to have major reduction in the use of real nuclear radioactive sources for training purposes.

In nuclear safeguards usually only traditional training methods such as books, slides and oral presentations are used in training surveillance concepts. These tools are characterized by the lack of interaction from the student, who is a mere receptor of information. On the contrary VR applications require the input from the trainee, creating a two way flow of information between the user and the system.

This paper focuses on the use of VR tools for training purposes in Nuclear Safeguards, specifically treating the subject of basic surveillance concepts.

A previous prototype [4] was the precursor of this work as it aimed at demonstrating the technological feasibility of such an application and the potential of VR methods. This paper introduces a VR tool ready to use for training basic surveillance concepts, its implementation characteristics, its testing at a real training course and future developments to be undertaken guided by the feedback obtained in the testing.

This application has been developed jointly by the EC-Joint Research Centre in collaboration with the IAEA within the framework of the European Commission Support Programme to the IAEA - Task EC-B-01876 - Development of Virtual Reality Tools for Safeguards Training.

## 2. Application objectives

The main target of this tool is complementing IAEA's Introductory Course to Agency Safeguards (ICAS).

This course intends to teach future IAEA inspectors the basic knowledge they will require to do their job in several topics.

One of these topics is video surveillance for Safeguards including fundamental concepts, such as the installation and the setup of surveillance cameras in Safeguards relevant areas.

The application intends to test the students' knowledge on the topic by presenting a realistic case she/he must solve using the knowledge acquired in the course.

To ease the learning process, the main goal is divided into series of sequential sub-objectives which need to be completed during the exercise. Each step highlights different single concepts.

As a secondary objective, this application provides a review tool for inspectors to determine the validity of a setup before actually trying it on the field, saving time.

## 3. Application characteristics

There are three separate key elements which compose this application. These are the **virtual model** (scenario, cameras and relevant objects), the **camera installation and setup procedure** and the **validation tools**.

### 3.1. Virtual model

This is one of VR's characteristic elements. The application provides a virtual model of a realistic scenario which an inspector might have to work in. In this application the scenario is a spent fuel storage site.

The model contains the usual elements which can be found in a spent fuel storage site, such as casks with nuclear fuel, entrances, a crane and others. Unlike static training elements like pictures or diagrams, these VR elements are interactive and the student can use them in the exercise showing in real time the effect of hers/his actions.

Also the surveillance camera model to be installed is modelled making it easier for the user to feel immersed in the virtual environment as if it were the real one.

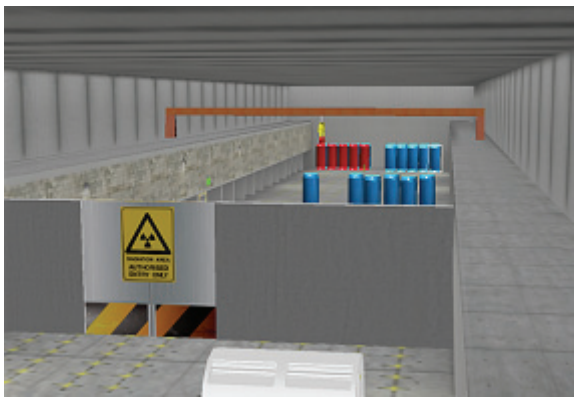

**Figure 1** Virtual Scenario

### 3.2 Camera installation and setup

This part of the application is where the knowledge of the user is tested.

There are two modes available;

a) the tour mode guides the user along the installation exercise step by step.
b) the expert mode, where there is no guidance and the user must solve the exercise by her/himself.

The user must complete a series of steps in order to correctly install and setup the surveillance camera(s). These steps require the user to perform an action in the virtual scenario which will result in real time feedback to the user. Indeed, the user can observe in real-time the result of the choices and settings made.

The tasks included in the application are:

(i) Placing a camera in an appropriate place in the scenario where it covers the key safeguards elements as the following figure shows.

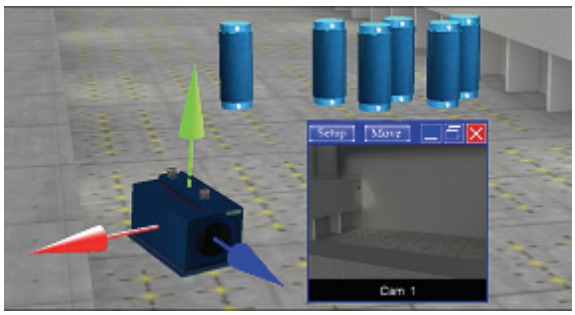


**Figure 2** Placing a camera in the scenario

(i) Subsequently setting up the parameters of the camera via an interactive setup menu which shows the resulting camera image in real time (Fig 3).

(ii) Finally evaluating the installation by using the tools available.

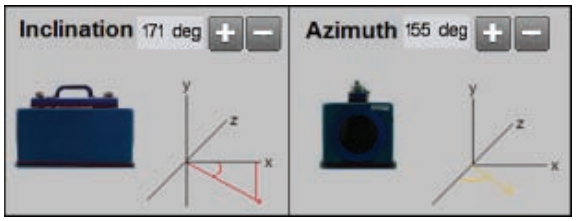


**Figure 3** Setting up the camera parameters

A checklist allows the user to view the tasks he/she has already performed and what is left to do in the exercise.

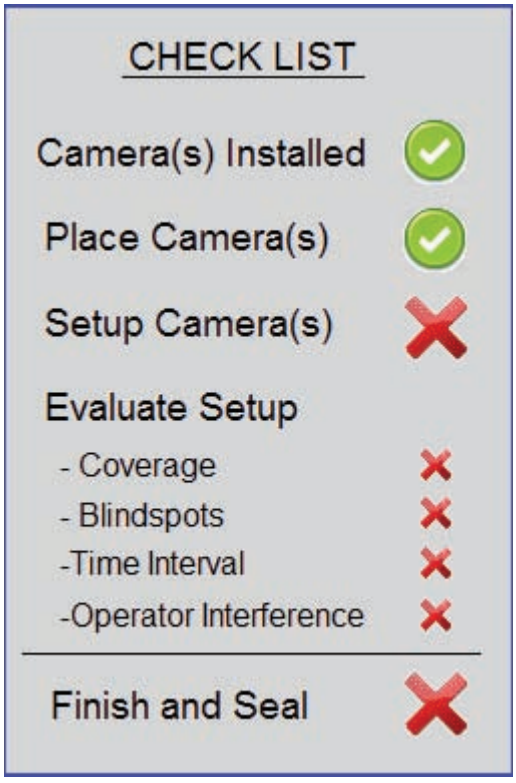


**Figure 4** Checklist shows progress on the exercise

### 3.3 Validation tools

The validation tools included in the application are a series of four functions:

(a) Coverage.
(b) Blindspots.
(c) Time Interval
(d) Operator interference.

These help the user evaluate the setup of her/his surveillance installation.

Thanks to VR visualization techniques, relevant information can be shown to the user in a graphical and easy to understand manner.

The first function, *Coverage*, checks if the safeguards relevant areas (in this exercise the two entrances) are covered by at least one of the cameras installed. If the installation does not comply with this requirement a warning message is displayed when the function is executed as shown in figure 5.


**Figure 5** Coverage test fail, warning message.

The second function, *Blindspots*, shows the volume of the scenario which is occluded to a certain camera by one of the casks. This helps the user to see if important areas might become out of sight or if there are points uncovered by all cameras. To pass the test at least one camera must be covering each entrance, otherwise a warning message is displayed.

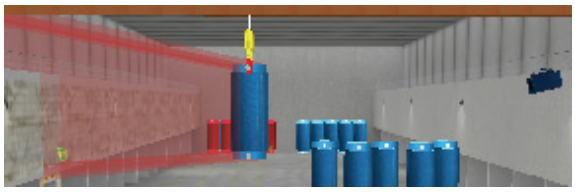

**Figure 6** Occluded volume to the camera (in red) show by Blindspots tool

The third function, *Time Interval*, checks that the Picture Taking Interval (PTI) of the camera is sufficient to distinguish the direction of movement of the transported cask, which might be important when analysing the flow of material subject to safeguards control. As with the other previously presented function, if the PTI is not correctly set a warning message will warn the user.

The last validation tool, *Operator interference*, allows the user to see if the standard movement of casks in and out of the installation by means of the crane may lead to a collision with the surveillance equipment installed.


**Figure 7** Main view of the application after having successfully created a surveillance setup with two cameras.

## 4. Application implementation

With the target of maximum portability of the application, ease of use and cost reduction the following technical implementation choices were made.

### 4.1 Development platform

For this work, 3DVIA Virtools was chosen as the development and deployment platform. This programming suite is specifically designed to create 3D Virtual Reality applications. Among other noticeable characteristics, it allows the programmer to import 3D models files from the most extended modelling software tools such as 3D Studio or Catia and neutral formats.
Furthermore it simplifies the use of VR hardware as it includes a series of plug and play drivers for an extensive list of VR specific hardware such as head mounted displays (HMDs) movement recognition devices.

### 4.2 Software requirements

Associated to the development software a web player is provided, this software tool allows to run the application on a web browser independently of the underlying operating system or other software technical details.

Therefore the only software requirement is a standard web browser (such as Mozilla Firefox or Internet Explorer) and the Virtools web player add-on.

These software tools are not only compatible with all common computer types but furthermore are free charge available via internet download.

### 4.3 Hardware requirements

With the aim of portability in mind this application is targeted to run on standard office PCs without any particular requirement.

The user interface hardware is reduced to mouse, keyboard and screen (preferably over 17”). More sophisticated devices such as HMDs were tested in the past without providing a practical advantage, in fact the extra cost and lack of comfort for the user represents a drawback for practical purposes.

This technical simplicity makes the package useful for refreshing courses (implying huge cost reductions) both for this application as well as for other VR based Safeguards training packages. The following figure shows the key map which highlights the keyboard interface controls of the application.

Other user input and visualization interfaces (e.g., head-mounted displays) can be easily coupled to the developed application.

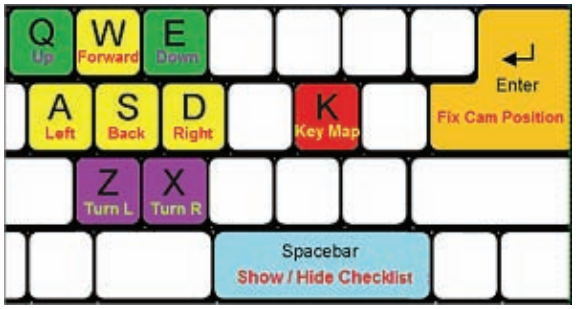


**Figure 8** Key map illustrates the keyboard interface description

## 5. Results and discussion

A testing session was scheduled to try out the application with IAEA users. The training session involved the participation of around twenty IAEA staff were divided in three groups according to the professional experience.

The subjects received a brief introduction to the application and were told to try out both modes of the application where a surveillance system needs to be installed in a spent fuel storage site with two entrances (see fig 1). Technical assistance was provided by the training staff when requested by the users.

Within the time slot of 45 minutes given for the testing session, all users successfully completed the exercise in both modes.

After the test the users were asked to fill in a feedback form with their impressions and suggestions about the tool.

## 6. Conclusion and Future work

The feedback from the test users was generally very positive. The testing session proved that a VR tool of this type is not only feasible but actually it can bring benefits to training methods in the field of Nuclear Safeguards.

The deployment of the application on different types of machines didn't create any problem highlighting the portability of the application.

Nevertheless, despite the successful test some limitations of the current version were identified. These are extremely useful as they represent the starting point to improve the application and develop a new version with new functionalities and improved performance.

## 7. Acknowledgements

The authors wish to thank Emilio Ruiz Morales and Damien Brasset who provided major contributions to the demonstration prototype of this training application. Special thanks also to Messrs. Matthew Heppleston (formerly with European Commission's DG-ENER) and Martin Möslinger (IAEA) for the discussions and input regarding the initial features of this simulation and training tool.

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# Continuous Can Content Monitoring for Special Nuclear Material Control and Accountancy at Sellafield

**Ludovic Bourva[1], Paul Caspall-Askew[2], Darius Ancius[3], Andreas Smejkal[3], Colin Wilkins[1] and Yves Lahogue[3]**

[1] CANBERRA UK Ltd.
Harwell Oxford, Building 528.10 Unit1,
Didcot, Oxfordshire, OX11 0DF, United Kingdom.
E-mail: ludovic.bourva@canberra.com, colin.wilkins@Canberra.com

[2] CE&I System Capability, Sellafield Ltd.
Hinton House,
Risley, Warrington, United Kingdom.
E-mail: paul.j.caspall-askew@sellafieldsites.com

[3] European Commissions,
DG Energy, Directorate Nuclear Safeguards,
1 Rue Henry Schnadt, Zone d'activités cloche d'or, L-2530, Luxembourg.
E-mail: Darius.ancius@ec.europa.eu, Andreas.smejkal@ec.europa.eu,
Yves.lahogue@ec.europa.eu

***Abstract:***

*A new facility has been designed to provide safe and secure storage of nuclear materials on the Sellafield site in Cumbria, UK. The facility opened in 2010 and received its first consignment of nuclear material in early 2011. The facility integrates the latest standards in terms of material storage safety and security and relies on two dedicated non-destructive assay systems for performing item verification measurements on cans of plutonium and mixed oxide (MOX) product prior to acceptance and inventory verification for nuclear safeguards purposes.*

*The Can Content Monitor (CCM) assay systems, developed by Canberra UK in close collaboration with Sellafield Ltd., and EURATOM to ensure safeguards by design, are custom designed instruments for the automated measurement of cans of plutonium or MOX. The monitors use neutron coincidence and multiplicity counting coupled with high resolution gamma spectrometry to determine the amount of plutonium and uranium present in each can. The monitors are expected to measure the complete input and output streams of the store.*

*The measurements require a high level of accuracy, and at the same time the monitors are also required to be fully automated. This latter requirement meant that a standard multiplicity counter design had to be adapted to allow automated loading and unloading of cans from/to the plant can transport system as well as a dedicated control and data interface to allow the plant control system to initiate measurement operations and for measurement data to be transferred to the plant database.*

*The design of the counter has also been adapted to allow remote independent Nuclear Safeguards systems, supplied by EURATOM, to monitor the detector signal chain outputs in order to safeguard the nuclear material within the plant. On the other hand, the design also allows for monitoring of the essential instrument parameters by the inspectors thus providing the required assurance for the validity of the measurements. A containment and surveillance scheme ensures the continuity of knowledge following the measurement.*

*This publication presents the overall philosophy, design, measurement functionalities, calibration approach and performance of these two monitors during their first full year of operation.*

**Keywords:** NDA system; store; automated system; multiplicity counter.

# 1 Introduction

A new plutonium storage facility has been built on the Sellafield site in Cumbria since 2005. The plant is designed to provide safe and secure long-term storage for the plutonium product resulting mainly from fuel reprocessing activities on site (THORP and Magnox oxides). It is also planned to store MOX residue from the Sellafield MOX Plant (SMP) and residues under various forms from older storage facilities on site.

The plant inactive commissioning was completed in 2010 and it received its first consignment of plutonium material in February 2011. Since then, the plant has been ramping up its activity to reach full operation during 2012.

To satisfy its Safeguards obligations under Article 77(a) of the EURATOM treaty the facility design integrates a Safeguards approach defined and to be applied by DG ENER.

In this regard, the interactions between the Safeguards Directorate and all the parties involved since the early design stages of the project represent a typical illustration of Safeguards by design [**1**]. Under the proposed approach most of the nuclear material stored at the facility will represent, once entering the plant, a static inventory. In order to establish a reliable material inventory, accurate quantitative measurements of the plutonium content of all of the nuclear material cans entering the facility is required. Such measurements also provide vital information to the operator to verify acceptance criteria of each item. Cans contents are similarly monitored when required by a store audit or for export to other plants. To support this material control and accountability process the facility commissioned Canberra UK (CUK) for the design and manufacture of two non-destructive assay systems. The supplied Can Content Monitors (CCM) hence represent key components for supporting the Safeguard Non Destructive Assay verification and accountancy tasks performed at the facility.

The two CCM counters, referred as CCM 209 and CCM 219, are designed to provide automated handling of the cans transported by the plant transfer system and to be operated by the plant Basic Control Plant System (BCPS). They support both gamma isotopic and neutron coincidence/multiplicity counting. The collected data is interpreted to derive the total quantity of special nuclear material in each can.

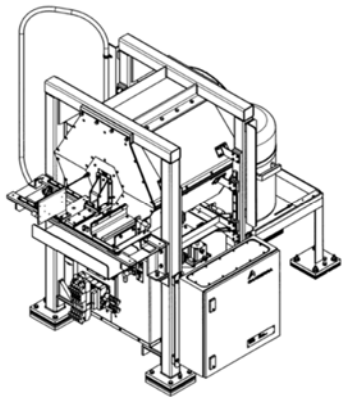

**Figure 1. 3D sketch of the non-destructive assay Can Content Monitor (CCM) systems.**

The remainder of this publication will present the design of the CCM counter and how it incorporates the plant operational and Safeguards requirements. The calibration approach used to produce the quantitative results is also reviewed and validation data collected over the first year of operation of the system on site is also presented.

# 2 CCM Systems Design

The CCM design is a typical example of a design combining the integration of standard nuclear instrumentation and specific plant operation/Safeguards requirements resulting in a unique purpose-built non-destructive assay system. The CCM counters combine both automated neutron and gamma acquisition chains based primarily on CANBERRA supplied standard instrumentation for special nuclear material assay as well as custom mechanical and software components which directly interface with the plant. Development and validation of such a system have therefore required unique capabilities in counter design as well as custom engineering and integration into the plant operation.

## 2.1 System Components and Functionality

Each monitor incorporates a can loading mechanism connected to trolley tracks that form part of the plant can transfer system. Cans are automatically loaded into an articulated neutron counting body made of two semi hexagonal High Density Polyethylene (HDPE) blocks. These house a total of 72 $^{3}$He gas proportional neutron counters and define the geometry of the measurement cavity. In order to perform plutonium isotopic

measurements, a collimated high resolution hyper pure germanium detector is fitted into one end of the counter to view the can inside the cavity along its axis. Because of the wide range of americium content in the measured cans, the system is also fitted with an automated gamma-ray attenuator assembly. Figure 2 shows the main components of the systems.

Figure 3 shows a high level block diagram illustrating the various detection and data acquisition components of the CCM.

The system uses a $^{252}Cf/^{133}Ba$ neutron/gamma check source, which is used during system standardisation/Quality Control measurements. The source is safely stored in a shield assembly and mounted on a retractable Teleflex cable which is used to move the source into the chamber during the standardisation count powered by an external drive mechanism.

All data acquisition electronics are located in a single nucleonics cubicle, installed in the vicinity of the counters.

In terms of operational constraints the large plutonium masses and range of materials types expected in this application presented complications to the system design.

- Gamma-Ray Exposure Rates would require the use of an internal shield to prevent gamma-ray pile-up events within the neutron counting chain.

The considered plutonium masses and high burn-up materials would result in very large total neutron emission rates, thus requiring the use of the latest technology

- neutron multiplicity shift registers.
- High gamma-ray exposure rates (primarily from $^{241}Am$ decay) from high burn-up materials require the use of an automated gamma ray attenuating filter mechanism

Also note that the system's design results from extensive MCNP [**2**] modelling of the chamber response to geometry adjustments and specific parameters like tube embedding depths or plug/shielding design which was performed to optimise the counter performances.

### 2.1.1 Can loading Mechanism

The counter has been designed as a hexagonal shaped split moderator/shield to facilitate the loading and unloading of the product cans from the can transport trolley mechanism. The upper shield is stationary while the lower shield raises / lowers approximately 15-mm by a pneumatic solenoid valve and air cylinder system. When loaded the can is translated on a bridging trolley track matching these used by the plant can transfer system. During load/unload operations the lower shield is positioned in the lowered position while it is in the raised position during the can measurements. Once the can is positioned within the counter, the track section assembly bridging the entrance of the cavity is translated sideways to position the front moderator plug in the alignment of the measurement cavity. The plug is then moved to close the cavity. Figure 4 shows a photograph of the can loading mechanism. With the lower shield raised, the assay chamber cavity interior dimensions are approximately 268-mm wide by 700-mm long.

The entire can loading process and the control of the built in standardisation source is controlled directly by the plant BPCS control system prior to the start of each measurement.

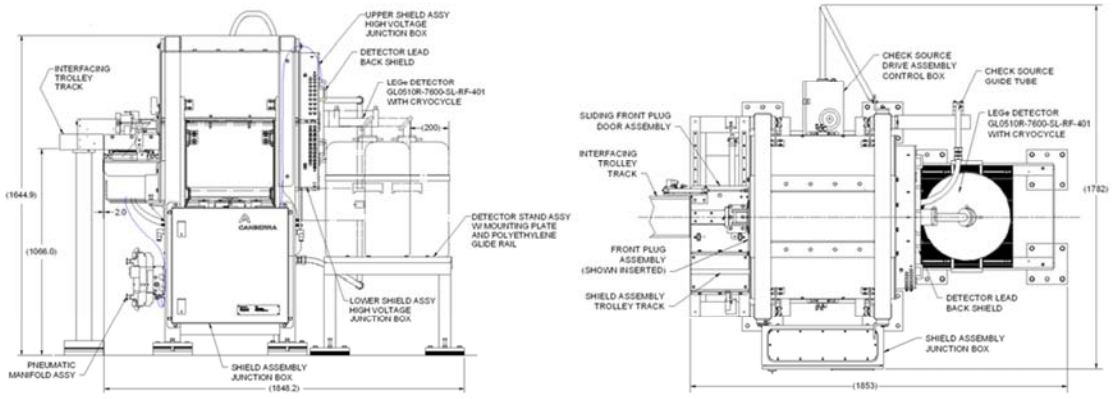


**Figure 2. Cross Section plots of the CCM showing the main components of the system**

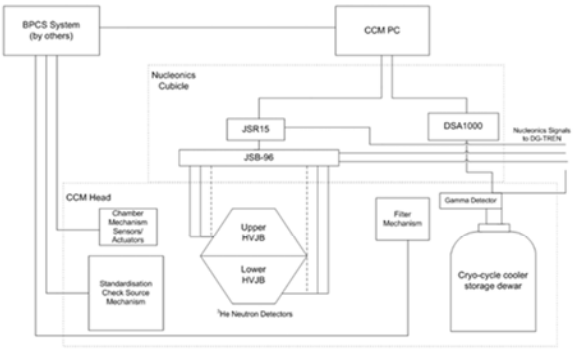


**Figure 3. Block component diagram for the CCM**

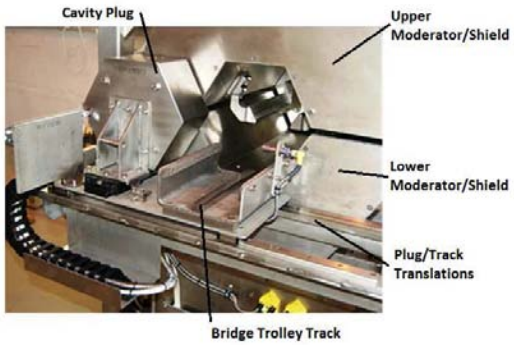


**Figure 4. Photograph of the CCM can loading mechanism**

### 2.1.2 Neutron Acquisitions

The main body of the counter is hexagonal shaped with three rows of 1" diameter $^{3}$He tubes (proportional counters) embedded in HDPE. Data collection is supported by two high voltage junction boxes (HVJBs) with 36 $^{3}$He tubes (609.6-mm active length and 10-atmosphere fill pressure) per junction box. This configuration yield for $^{240}$Pu spontaneous fission neutrons a detection efficiency of the order of 45 %. Figure 5 illustrates the geometry of the neutron counters.

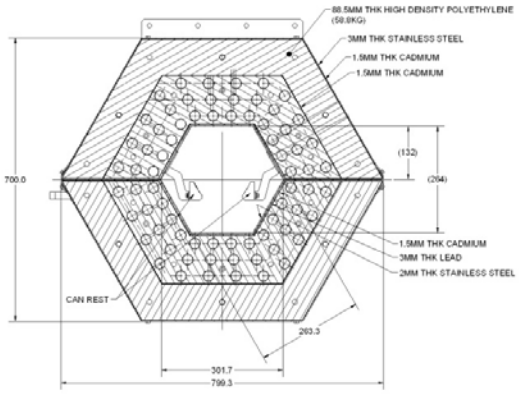


**Figure 5. Cross section plots of the CCM showing the neutron detector layout in the counter main body**

The detectors for each HVJB are wired to provide 18 separate counting chains. Each pair of tubes is ganged together through a single JAB-01 Amplifier/Discriminator Board to form a single counting channel. The JAB-01 is a Canberra manufactured amplifier/discriminator board that has an analogue and digital section with separate, unconnected ground planes. Components on the JAB-01 include nine integrated circuits, the AMPTEK A-111 hybrid amplifier/discriminator, a 74221 dual one-shot, and a 74S140 dual 4-input 50-ohm line driver. The threshold of each A-111 is adjusted to match the gain and minimize the response to gamma rays. The JAB-01 board outputs are 52-nsec width TTL pulses that are fed into a JSB-96 summer/derandomiser unit located in the radiometric enclosure. The JSB-96 generates two signal chains:

- One signal combining the output from all detectors. This full efficiency neutron counter pulse train output is fed to a CANBERRA type JSR-15 neutron multiplicity shift register.
- Second signal combining the outputs of a lower efficiency subset of the detectors (individual rings). These low efficiency neutron detector pulse train outputs are not used directly by the CCM but are provided for compatibility with shift registers whose maximum count rate capability may be exceeded when high mass high burn-up materials are present in the chamber.

### 2.1.3 Gamma Acquisitions

The Gamma detection system uses a high purity germanium detector (HPGe). The detector is a CANBERRA GL0510 Low Energy Germanium (LEGe) type with RC pre-amp. The detector output is analysed by a CANBERRA DSA-1000 Digital Signal Processor. The GL0510 detector is the

standard detector model used with the U/Pu Inspector (known as the IMCG system).
The gamma detector is mounted in a 7600 horizontal dipstick configuration and 90mm extended cryostat length along with RFI and Cryo-Cycle options.
The gamma attenuator assembly is provided to accommodate the expected range in gamma-ray emission rates from the various sample types. The attenuator assembly consists of two tin filters that can be retracted or extended in front of the gamma detector. Both attenuators are located between the detector end-cap and the stainless steel end-plug. Each attenuator sheet is connected to a BPCS controlled pneumatic solenoid valve and cylinder. The positions of the two attenuator actuators are monitored by four sensors.

## 2.2 System Operation

### 2.2.1 Plant Operation

Standard CCM operation is performed in an "Operational" mode for which the system supervisory program is interfaced with the plant BCPS. The interface between the CCM software and the BPCS uses the OPC Data Access protocol. Practically this allows a plant operator to initiate from the BCPS all CCM measurements. The CCM plant operation also includes two maintenance modes, which allow further control and diagnostics on the assay parameters are available. Routine operation supported by the CCM system includes the measurement of ambient background, standardisation sources and calibration check cans. These operations must be completed prior to measuring a specific can.
To perform a can assay the BPCS must load the can into the measurement chamber and ensure that the standardisation sources are shielded. The BPCS should then transmit data for the can to be assayed to the CCM software. On receipt of the data and a start assay signal from the BPCS the CCM makes the following verifications:

- System in set in "Operational" mode.
- No measurement is in progress.
- The Can data has been received from BPCS.
- The Standardisation results are valid.
- The Background results are valid.
- The Calibration Check Can results are valid.

If all checks are successful then the CCM through a supervisory program will manage the assay. The CCM software first requests CANBERRA NDA2000 [**3**] to carry out a pre-measurement in order to determine the optimum attenuator setting for the HRGS detector, having first instructed the BPCS to set the attenuator to its starting position. The HRGS detector count rate observed during this short pre-count is used to determine whether the attenuator setting should be changed, and if so, the process repeated until the optimum count rate is achieved. The CCM software will then instruct NDA2000 to start the gamma/neutron assay. The CCM assays perform the measurement and analysis of the high resolution gamma spectrum as well as the acquisition of neutron coincidence/multiplicity count rates. Data acquisition is performed over 900 seconds, with the neutron data acquired in 30 cycles of 30 seconds each. Background rejection is applied on the neutron cycle data.
The determination of the plutonium isotopic composition relies on the Multi-Group Analysis (MGA) algorithms [**4**] in its version 9.63H, as supplied by CANBERRA as part of its Genie 2000 gamma spectroscopy software suite. The measured plutonium isotopic is used in the passive neutron coincidence data analysis to derive the $(\alpha,n)$ to spontaneous fission neutron emission fraction, $\alpha$.
The measured background corrected, dead time corrected coincidence count rate, inferred from the raw cycle rates, are used to derive, using the measured $\alpha$-value, the multiplication corrected Reals count rate [**5**]. This rate is interpreted as a $^{240}$Pu effective mass ($^{240}Pu_{eff}$) using the established calibration.
The Single, Double and Triple coincidence count rates are also provided by the JSR-15, and the $^{240}Pu_{eff}$ mass is therefore also directly derived from these rates and the multiplicity equations [**6**]. However, the very high count rate recorded for this application does impact on the precision of the multiplicity results. Therefore neutron multiplicity results, although considered as more accurate than the PNCC results, are not used for mass reporting, but provide secondary validation on the validity of the assumptions built in the PNCC analysis (sample purity, uniform isotopic composition…).

### 2.2.2 Safeguards Measurements

There are four signals being provided from the CCM nucleonics for use by Safeguards systems operated by EURATOM (DG-ENER) to monitor both neutron and gamma counting signals and the PNCC HV supply output setting. This is illustrated in Figure 6.

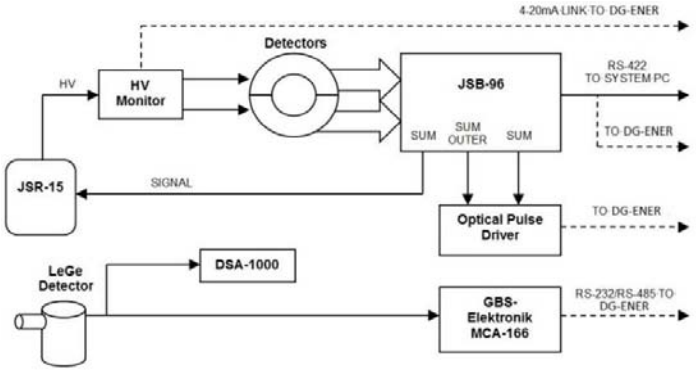


**Figure 6. Illustration of the architecture used to provide signals to the Safeguards systems**

The two neutron summed pulse train output signals (Total and Ring outputs) provided by the JSB96: are transmitted to the EURATOM system along optical fibres using a Canberra type 950260-1 optical pulse driver. The optical pulse driver converts two TTL signal inputs from the JSB-96 to two optical signal outputs for transmission by fibre-optic cable to the EURATOM system, where the signals are acquired by EURATOM's JSR-15. The Gamma-ray isotopic detector preamplifier energy output is also sent to the EURATOM system. The detector preamplifier is provided with an additional identical signal output. This is fed to the EURATOM supplied MCA-166 MMCA [**7**]. The MMCA and JSR-15 both are controlled from a remote external location by a EURATOM supplied PC via a RS-232/RS-485 serial interface. For the automatic acquisition of the data, Euratom uses their standard RADAR software, the analysis is performed with CRISP [**8**].

Finally the JSR-15 HV output monitoring is carried out using CANBERRA type 950446-1 High Voltage (HV) Monitor. The HV Monitor incorporates a high voltage and high impedance potential divider with a 4-20mA current loop output, which is made available via an isolation amplifier. The input side of the isolation amplifier is powered from the JSR-15 so that a common ground reference is established. This allows for the HV to be monitored to +/- 10 volts.

The serial communications traffic between the CCM and the JSB-96 is also monitored from an external location by a EURATOM supplied listening device. The serial interface from the JSB-96 to the CCM PC is split using a proprietary 'Tap' connector to provide DG-ENER with a duplicate read-only signal, enhancing EURATOM's capabilities to check the correctness of the system settings.

# 3 NDA Systems Calibration

As part of the system's commissioning extensive testing and characterisation of the CCM response was carried out at the CANBERRA UK factory in Harwell, UK. These test measurements allowed the determination of the operational characteristics and settings of the counter's nuclear instrumentation and ensures that the systems are used according to best practices for nuclear material assay [**9**].

## 3.1 Factory Calibrations

Once basic system testing and characterisation was performed, a set of fully characterised plutonium oxide standards [**10**] were used to determine the coincidence response as a function of $^{240}Pu_{eff}$ mass for two can types. The set consists of 10 sources in the $^{240}Pu_{eff}$ mass range 0.02g to 5.6g. Although this set of sources did not provide a mean to exercise the counters in a regime similar to the one to be encountered on site, it provided a reliable preliminary calibration. The plutonium and americium data were revised to a specified measurement dates.

The calibration process determined the relationship between the $^{240}Pu_{eff}$ mass and the measured neutron coincidence rate. Each of the standards and combination of standards were positioned in the inner and outer can assembly using purposed built source holders. Each sample was counted for 120 repeat measurements of 30 sec duration (1 hour total counting time). The input data to the calibration routine were hence sets of data points ($^{240}Pu_{eff}$ mass (grams) and the background and dead time corrected count-rates) from which the calibration curves, shown in Figure 7, were constructed.

The calibration parameters for the multiplication corrected Reals rate of THORP and MAGNOX products, derived from the fits shown Figure 7, are presented in Table 1.

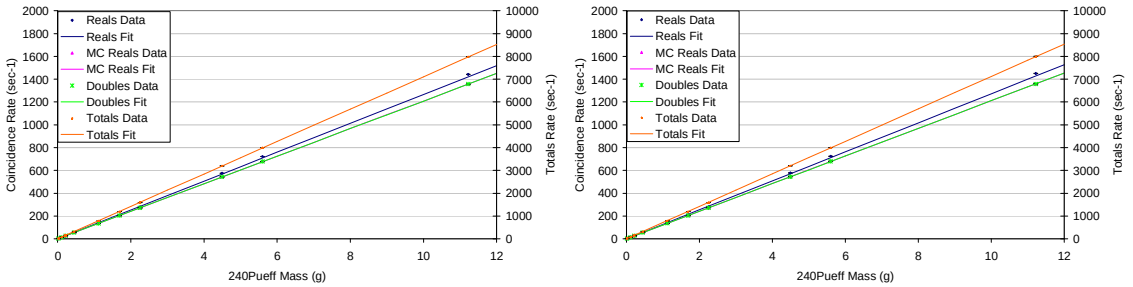


**Figure 7. Empirical calibration data performed on the CCM for THORP (LHS) and MAGNOX (RHS) material**

**Table 1. CCM PNCC factory calibration coefficients for the multiplication corrected Reals rate**

| | Multiplication Corrected Reals Rate Calibration Coefficients | |
|---|---|---|
| | **MAGNOX can** | **THORP can** |
| **Slope $g_1$ ($s^{-1}.g^{-1}$)** | 121.160 | 120.88 |
| **Offset $g_2$ ($s^{-1}.g^{-1}$)** | 0 | 0 |

## 3.2 Site Calibration Validation Measurements

Following plant installation of the CCM systems, a number of validation measurements were carried out on THORP product cans (48-assays), MAGNOX product cans (54 assays) and SMP Residue cans (52-assays). These measurements served as a baseline to establish updated calibration parameters for the multiplication corrected Reals rate analysis. Re-analysis of the measured neutron coincidence count rates was performed using supplied reference isotopic data for the assayed cans in order to derive the material reference alpha value. This highlighted the need to improve the accuracy of the isotopic measurements, in particular with regards to the accuracy of the isotopic correlation used to derive the $^{242}$Pu fraction. Additionally, as CANBERRA UK did not have access to MOX sample during the initial calibration effort described in Section 3.1, a specific SMP residue material calibration was derived as part of this data review. Consequently an SMP calibration for these assays was introduced in NDA2000. This required creating a new sample type for SMP Residue linking it to the SMP Residue calibration data for the SMP Residue count type.

The plant operation meant that the CCM 219 counter had mainly measured on THORP/SMP cans while CCM 209 mainly assayed MAGNOX cans. Consequently, it has not been possible to date to establish a MAGNOX calibration curve specific to the CCM 219, or THORP/SMP calibration curve specific to the CCM 209. Calibration parameters established at CUK for similar samples on the two counters having shown that they have relatively close performances.

It has therefore been proposed, until further data is available, to consider data set obtained on either counter to be transferable as “best estimate” for the other chamber.

Re-analysis of the series of THORP, SMP cans (21 assays) with the updated parameter showed good consistency with the reference data supplied by the plant. This is illustrated in Figure 8, where a plot of the measured total plutonium against reference total plutonium data for SMP and THORP material measured on CCM 219 is shown. Agreement between measured and reference plutonium values averaged 0.12 % with a standard deviation of 1.4 %.

**Table 2. CCM PNCC On-site Calibration Parameters for the MC Reals Calibration Curve**

| Count Type | Container Type | Sample Type | Slope Coefficients | | Offset Coefficients | |
|---|---|---|---|---|---|---|
| | | | Slope | Var | Offset | Var |
| THORP Product | THORP Can | THORP Pu | 121.416 | 0.004588 | 0 | 0 |
| MAGNOX Product | MAGNOX Can | MAGNOX Pu | 118.108* | 0.000660* | 0 | 0 |
| SMP Residues | THORP Can | SMP Pu | 124.479 | 0.009488 | 0 | 0 |

*** Value determined from CCM 209 Magnox Can Assays.**

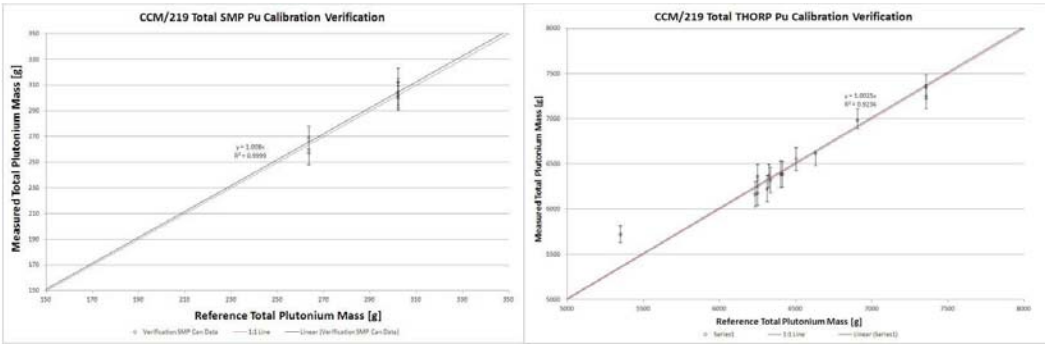


**Figure 8. CCM 219 comparative plot between reference and measured plutonium content in SMP (LHS) and THORP (RHS) cans**

During the re-analysis of THORP, MAGNOX and SMP raw data acquired at the plant, it was shown that isotopic analysis using the MGA v9.63H software returned inaccurate evaluation of the $^{242}$Pu fraction. In-accuracy in the determination of the $^{242}$Pu fraction lead to biased alpha value attributed to the measured material and, in the case of a multiplication corrected Reals analysis to biases between derived $^{240}Pu_{eff}$ and total Pu masses. This is illustrated in Figure 9 where plots of the declared vs. measured quantities are shown for $^{240}Pu_{eff}$ (LHS) and total Pu (RHS).

To correct for this an updated version of MGA (v10.0) has been installed on the CCM systems. This version supports the custom parameterisation of the isotopic correlation algorithm used to derive the $^{242}$Pu fraction. A study has been conducted to derive improved parameter for the CCM measurements. To date the implementation of this new "custom" algorithm has not been performed as this will require a software modification at the level of the CCM software.

## 4 Conclusions

Two fully automated non-destructive assay systems were installed at the Sellafield in 2010. These two systems provide essential support to plant operation as well as safeguard verification activities. Their design represents a unique blend of standard nuclear instrumentation and custom specification to provide a fully integrated measurement solution. System's calibration performed prior to delivery of the system and verification measurement based on larger and more diverse batch of material at the facility have shown good agreements and confirmed the measurement capabilities of the systems.

## 5 Acknowledgements

We would like to acknowledge Peter Chare from DG-ENER for his strong involvement and contribution to the project. Our thanks go as well to Peter Schwalbach, also from DG-ENER, for reviewing this manuscript.

## 6 References

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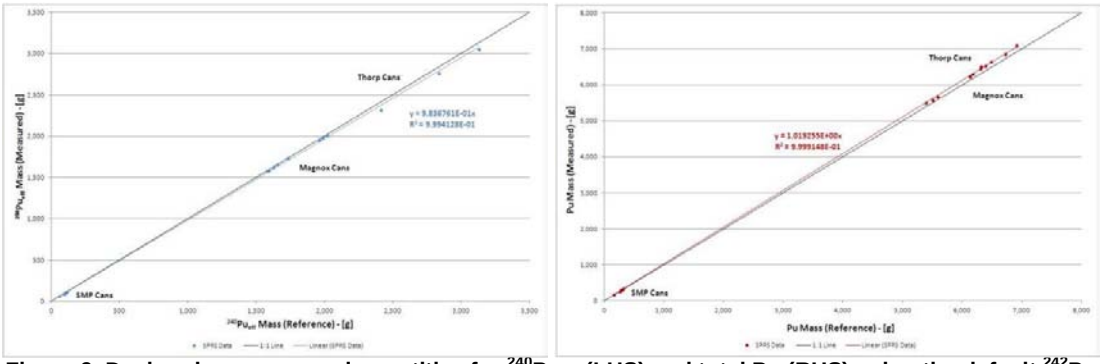


**Figure 9. Declared vs. measured quantities for $^{240}Pu_{eff}$ (LHS) and total Pu (RHS) using the default $^{242}$Pu isotopic correlation setting in MGA v9.63H**

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# Collection and Analysis of Open Source News for Information Awareness and Early Warning in Nuclear Safeguards

**Giacomo G.M. Cojazzi, Erik van Der Goot, Marco Verile, Erik Wolfart**

Joint Research Centre, European Commission
Via Fermi, Ispra 21020 (VA) Italy

**Marcy Rutan Fowler, Yana Feldman, William Hammond, John Schweighardt, Matthew Ferguson**

International Atomic Energy Agency
Wagramerstrasse 5, P.O. Box 100, 1400 Vienna, Austria

***Abstract:***

*Acquisition and analysis of open source information plays an increasingly important role in the IAEA's move towards safeguards implementation based on all safeguards relevant information known about a State. The growing volume of open source information requires the development of technology and tools capable of effectively collecting relevant information, filtering out "noise", organizing valuable information in a clear and accessible manner, and assessing its relevance.*

*In this context, the IAEA's Division of Information Management (SGIM) and the EC's Joint Research Centre (JRC) are currently implementing a joint project to advance the effectiveness and efficiency of the IAEA's workflow for open source information collection and analysis. The objective is to provide tools to support SGIM in the production of the SGIM Open Source Highlights, which is a daily news brief consisting of the most pertinent news stories relevant to safeguards and non-proliferation. The process involves the review and selection of hundreds of articles from a wide array of specifically selected sources.*

*The joint activity exploits the JRC's Europe Media Monitor (EMM) and NewsDesk applications: EMM automatically collects and analyses news articles from a pre-defined list of websites, and NewsDesk allows an analyst to manually select the most relevant articles from the EMM stream for further processing.*

*The paper discusses the IAEA's workflow for the production of SGIM Open Source Highlights and describes the capabilities of EMM and NewsDesk. It then provides an overview of the joint activities since the project started in 2011, which were focused i) on setting up a separate EMM installation dedicated to the nuclear safeguards and security domain (Nuclear Security Media Monitor, NSMM) and ii) on evaluating the NSMM/NewsDesk for meeting the IAEA's needs. Finally, it presents the current use NSMM/NewsDesk at the IAEA and proposes options for further integration with the IAEA workflow.*

**Keywords:** open source information, information collection and analysis, media monitoring, non-proliferation, nuclear safeguards

## 1. Introduction

Open source information is defined as any information that is neither classified nor proprietary. It includes but is not limited to: media sources, government and non-governmental reports and analyses, commercial data, and scientific/technical literature. Open source information is one of several types of information that is used by the IAEA Department of Safeguards to verify States' compliance with their safeguards obligations.

The growing volume of open source information, the diverse forms of information available (text, audio, visual), and the technical competence required to evaluate this information poses significant challenges to effective information collection and analysis. Meeting these challenges requires the development of technology and tools capable of effectively collecting relevant information, filtering out "noise", organizing valuable information in a clear and accessible manner, and aiding in the assessment of the information's relevance.

To improve the IAEA's ability to utilize open source information, a joint project is currently underway as part of the European Commission's (EC) support to the IAEA Department of Safeguards (task EC-D-1880) to optimise existing processes involved in the daily collection of open source information by the Global Monitoring Team (GMT) and production of SGIM Open Source Highlights (OSHL) [1]. The project utilizes the Europe Media Monitor EMM/NewsDesk application, developed by the EC's Joint Research Centre (JRC).

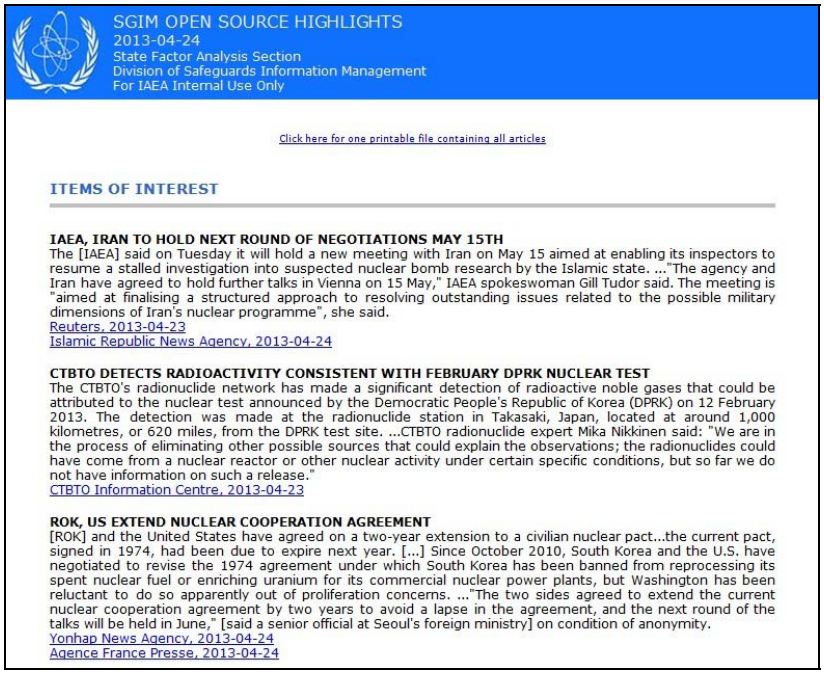
This paper describes the continuation of a project which was previously reported on at the Institute of Nuclear Materials Management (INMM) Annual Meeting in 2011 [2]. It describes the current workflow for the daily collection of open source information by the SGIM GMT, the production of SGIM Open Source Highlights, and the EMM and NewsDesk capabilities. Additionally, the paper discusses the testing that took place in the pre-launch phase (coverage tests), and the deployment and outcomes of the Phase 1 launch of the application (usability tests). Lastly, the paper details the ongoing joint activities to streamline the production of the OSHL and proposes efforts to enable a more effective and efficient mining of open source information to enhance state evaluation through continuous monitoring.

## 2. Open Source Information at IAEA

Acquisition and analysis of safeguards-relevant open source information plays an important role in the understanding of a State's nuclear fuel cycle and activities. All safeguards relevant information available to the IAEA is used not only to support the drawing of safeguards conclusions, but also to plan the optimal set of safeguards verification activities to be conducted in a State. The evaluation of all information is an ongoing and iterative process, in which knowledge about a State will continue to expand and improve. This allows the Agency to draw safeguards conclusion in the most effective and efficient way [3].

The Department of Safeguards uses open sources in several ways in support of the State evaluation process, including: i) daily review of open source information by the GMT to maintain current awareness and serve as early warning for safeguards-significant events; ii) as part of continuous monitoring of safeguards-relevant activities within a State, which contributes to a State evaluation report; iii) for focused research to support in-field verification activities such as complementary access; and iv) for special investigations related to a specific State, entity, fuel cycle step, or technology.

The IAEA's focal point for the collection and analysis of open source information is the Division of Information Management (SGIM) within the IAEA's Department of Safeguards. As part of its mandate for reviewing and analysing open source information to maintain current awareness and provide early warning for safeguards-significant events, SGIM produces a daily news briefing, called SGIM Open Source Highlights (OSHL), which covers safeguards-relevant news and analyses items, as well as relevant developments in nuclear non-proliferation, disarmament, arms control, and the nuclear fuel cycle. OSHL generally consists of four to eight news stories daily, and has a distribution list of nearly 600 staff members within the Department of Safeguards and other IAEA staff. A typical snapshot of SGIM Open Source Highlights is shown in Figure 1.



SGIM OPEN SOURCE HIGHLIGHTS
2013-04-24
State Factor Analysis Section
Division of Safeguards Information Management
For IAEA Internal Use Only

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ITEMS OF INTEREST

**IAEA, IRAN TO HOLD NEXT ROUND OF NEGOTIATIONS MAY 15TH**
The [IAEA] said on Tuesday it will hold a new meeting with Iran on May 15 aimed at enabling its inspectors to resume a stalled investigation into suspected nuclear bomb research by the Islamic state. ..."The agency and Iran have agreed to hold further talks in Vienna on 15 May," IAEA spokeswoman Gill Tudor said. The meeting is "aimed at finalising a structured approach to resolving outstanding issues related to the possible military dimensions of Iran's nuclear programme", she said.
Reuters, 2013-04-23
Islamic Republic News Agency, 2013-04-24

**CTBTO DETECTS RADIOACTIVITY CONSISTENT WITH FEBRUARY DPRK NUCLEAR TEST**
The CTBTO's radionuclide network has made a significant detection of radioactive noble gases that could be attributed to the nuclear test announced by the Democratic People's Republic of Korea (DPRK) on 12 February 2013. The detection was made at the radionuclide station in Takasaki, Japan, located at around 1,000 kilometres, or 620 miles, from the DPRK test site. ...CTBTO radionuclide expert Mika Nikkinen said: "We are in the process of eliminating other possible sources that could explain the observations; the radionuclides could have come from a nuclear reactor or other nuclear activity under certain specific conditions, but so far we do not have information on such a release."
CTBTO Information Centre, 2013-04-23

**ROK, US EXTEND NUCLEAR COOPERATION AGREEMENT**
[ROK] and the United States have agreed on a two-year extension to a civilian nuclear pact...the current pact, signed in 1974, had been due to expire next year. [...] Since October 2010, South Korea and the U.S. have negotiated to revise the 1974 agreement under which South Korea has been banned from reprocessing its spent nuclear fuel or enriching uranium for its commercial nuclear power plants, but Washington has been reluctant to do so apparently out of proliferation concerns. ..."The two sides agreed to extend the current nuclear cooperation agreement by two years to avoid a lapse in the agreement, and the next round of the talks will be held in June," [said a senior official at Seoul's foreign ministry] on condition of anonymity.
Yonhap News Agency, 2013-04-24
Agence France Presse, 2013-04-24

**Figure 1:** Snapshot of SGIM Open Source Highlights. News stories are presented in order of importance to safeguards activities.

## 2.1. Daily Review Procedures

The production of OSHL begins with a coordinated daily review by the Global Monitoring Team (GMT) of news repositories and websites. Sources of information include subscription-based news databases and aggregators, general news media sites, blogs, non-governmental organizations (NGOs), universities and research institutes, and government agency websites.

The daily monitoring consists of a preliminary manual review by the GMT of approximately 1500 news items on average. Prior to the Phase 1 NewsDesk deployment (see section 4.4), the daily monitoring consisted of three streams of information: i) a subscription-based news database, ii) website-tracking software set up to monitor over 100 news and analysis sites not covered by the database, and iii) a set of websites composed of nuclear-focused news aggregators, internal sources, and subscription-based publications.

Approximately 100 articles are selected for secondary review by the OSHL editorial team, which then selects articles either for inclusion in OSHL or for further back-end processing ensuring the long-term availability of the information. Following these previous procedures, the preliminary review required approximately three staff-hours, followed by the secondary review by the OSHL editorial team of approximately one staff-hour.

## 2.2. Production and Distribution of SGIM Open Source Highlights

Once news stories are selected for inclusion in OSHL, they are distributed to the OSHL editorial team. Editors research the stories, and create an abstract of the issue of approximately 3-5 sentences, pulling together several sources as necessary. The abstracts are posted as drafts in a blog format on a SharePoint intranet portal called the IAEA Safeguards Portal. The abstracts are then reviewed by an editor and published on the Safeguards Portal. Each abstract includes a link to a PDF of the full-text article(s) that is saved for archival purposes.

Once abstracts of all news stories for the day are completed, they are also distributed in an email newsletter to nearly 600 IAEA staff members. Production of the OSHL takes approximately an additional four to six staff-hours a day, depending on the volume of news and the complexity of the issues.

### 2.3 Back-End Processing

Because of the dynamic nature of information contained in websites, to ensure that the information that is identified as relevant by the GMT is available to the Agency in the future the daily monitoring of open source information also serves to build an IAEA internal knowledge base. Selected articles are manually processed to add metadata and then transferred to the internal repository. This process takes approximately one staff-hour per day. The resulting repository is used as a starting point for subsequent analysis work related to State evaluations and other safeguards verification activities.

## 3. Europe Media Monitor Platform and Tools

The Europe Media Monitor (EMM), developed by JRC [4], is a web-based multilingual news aggregation system that collects over 170 000 news articles per day in about 50 languages from more than 4000 web news sources. The sources are mainly general news sites with a world-wide coverage, but also include some specialist websites and twenty commercial news providers. The system employs text mining techniques to provide a picture of the present situation in the world as conveyed in the media. These techniques include automatic multilingual categorization, entity extraction, geo-location, quote extraction and sentiment analysis. In addition, an algorithm for detecting breaking news automatically clusters all collected news articles every ten minutes and displays the ten largest clusters per language by plotting them on a time-by-size graph. It also provides all the necessary hyperlinks to navigate through the clusters and to go to the source for a detailed exploration. Furthermore it applies some deeper semantic information analysis techniques, for example, to automatically detect violent events, derive reported social networks and analyze media impact [5].

EMM creates a searchable full-text index of all articles that flow through the system. For each article, it stores meta-information including title, description, source, category, language, and original URL (Uniform Resource Locator). However, it does not store the original article itself.

The EMM engine is a processing chain of lightweight, independently running processing modules. They were developed by the JRC as basic but very reliable web services communicating over standard HTTP (Hypertext Transfer Protocol) posts. Articles flow through the processing chain as thin RSS (Really Simple Syndication) items that grow as meta-data gets added at each stage of the processing chain.

Several EMM installations have been set-up providing varying thematic focus and user accessibility. The main EMM installation monitors generic news media with little coverage of specialised thematic areas and serves as a general media monitor and demonstrator of EMM capabilities. Its front page – the EMM Newsbrief [6] - provides a user interface to all this information and is visited on a regular basis by some 25 000 users, and gets some 1.5 million hits per day. A snapshot of the EMM website is shown in Figure 2. Other EMM installations are targeted at specific thematic areas and/or customers. For example, MedISys is specialized in medical and health-related topics [7].


**Figure 2:** Snapshot of EMM Newsbrief page. Current top stories are shown in the centre column. Access to thematic categories is provided through the links in the left column. All information is generated fully automatically.

### 3.1. NewsDesk

EMM is a powerful tool for automatically aggregating and analysing open source information from the Internet. However, if the information needs to be disseminated further or fed into an existing information analysis workflow, the generated news stream has to go through a review and selection process carried out by a domain expert. For this purpose, the JRC developed the NewsDesk application (see Figure 3).

NewsDesk allows for manual review and selection of the most relevant articles with an easy-to-use drag-and-drop interface. It also supports the rapid production of newsletters which can be disseminated to interested user groups. The articles selected in NewsDesk including the meta-information extracted by EMM can be posted to existing third-party IT systems, for example, to ingest the information into a back-end archive or to publish the selected news on a web portal.

NewsDesk is conceived as a collaborative environment, i.e. users are organised in virtual groups where they can work as a team on the news articles review and newsletter production. It is a web application, which uses RSS feeds as information input – typically generated by EMM. However, it is also possible to ingest feeds from third-party applications. NewsDesk allows users to send notifications via SMS or e-mail; it also integrates an automated notification system to alert personnel on duty during holidays or out of office hours.

### 3.2. Other EMM Tools

Besides NewsBrief web site and NewsDesk moderation tool, the EMM platform provides additional tools including:

- *Category Editor*: '*Categories'* are a fundamental part of the EMM system, as they allow generating automatic alerts and filters according to thematic or geographic areas of interest, thus significantly reducing the information overload. The Category Editor is a web-based application that enables users to create and maintain the category definitions, which can be based on complex keyword or meta-data combinations. A description of the category mechanism and results for applying it to the medical and health domain is given in [8].
- *Media Impact* is a media analysis tool that, starting from a dataset (typically news articles selected via a search in NewsDesk), allows media analysts to further tag and aggregate articles in order to produce statistics and reports. The primary purpose is to analyse how specific events are reported

in the world-wide media and to examine their media impact according to country, media type or other criteria. The tool exploits the meta-information automatically extracted by EMM together with manually added metadata.

- *EMM Mobile Apps* for both iOS and Android platforms addresses the need of a personalized access to the EMM content and functionality. The web site content selection is fully automated and its ergonomics offers a common navigation through the daily news. NewsDesk and Media Impact are meant for human-moderated news selection where final clients receive a product conceived and structured by analysts; EMM Mobile Apps ultimately gives end-users direct control on content selection and visualization in order to render their own view of the daily news [9].

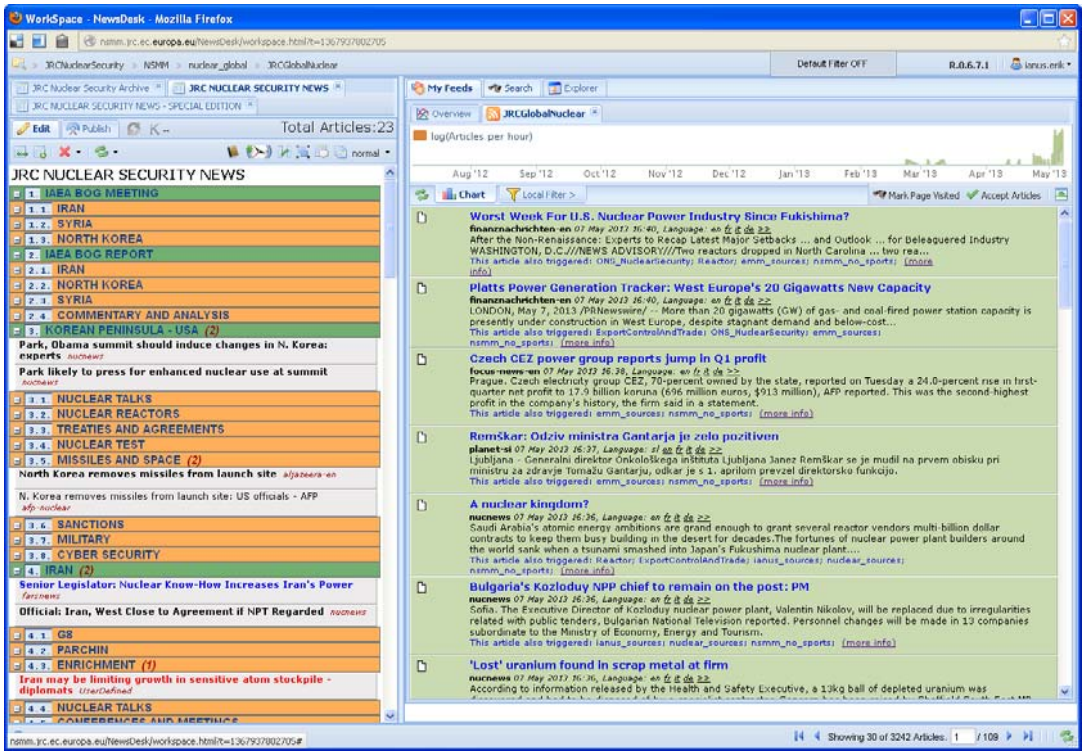


**Figure 3:** Screenshot of the NewsDesk workspace. The news feed generated by EMM is displayed on the right. The analyst reviews the information and drags relevant articles to the newsletter area. Selected articles are collected in the final newsletter on the left-hand side.

## 4. Optimizing IAEA's Open Source Collection Process

The IAEA intends to increase the efficiency of its open source information monitoring process and OSHL production workflow and therefore is interested in using EMM/NewsDesk. IAEA and JRC are engaged in a joint project to evaluate EMM/NewsDesk for IAEA needs and to analyse how the applications can be integrated in the existing IAEA workflows and systems.

The IAEA has a well-established workflow for the open source information collection and SGIM Open Source Highlights production, which is constrained by various boundary conditions including legacy IT systems, confidentiality considerations and time and human resource availability. The IAEA/JRC collaboration on the project aims to analyse EMM/NewsDesk compatibility with respect to i) source coverage ii) interoperability with existing IAEA back-end IT systems and iii) NewsDesk functionality and compatibility with IAEA workflow. It is anticipated that at a minimum EMM/NewsDesk will be used to process the information sources that IAEA currently monitors manually. Potentially, it can cover the entire workflow including the review of all sources, the production and dissemination of OSHL and back-end processing.

The remainder of this section illustrates the issues that have been addressed since the start of the project in order to introduce EMM/NewsDesk to the IAEA workflow and describes the current status of the activities.

## 4.1 Source Coverage

The public version of EMM monitors pre-selected websites targeting mainly general news media with little coverage of specialised thematic areas. Typically, important news stories are duplicated across many media sites, thus generating some redundancy in the system. Consequently, the system tolerates a certain degree of undetected articles, meaning that not all relevant articles of all targeted sites need to be retrieved.

The IAEA, however, also needs to monitor a set of specific nuclear safeguards-related websites, including NGOs, blogs and sites of national and international authorities. Hence, IAEA provided a list of more than 140 nuclear-specific websites, which it would like to be added to the EMM source list (hereafter referred to as nuclear websites).

In order to fine-tune EMM to IAEA needs without interfering with the public EMM website, it was decided to set-up a separate EMM installation dedicated to the nuclear safeguards and security domain, hereafter referred to as Nuclear Security Media Monitor (NSMM). The types of websites of potential interest to the IAEA and therefore monitored by NSMM include: nuclear-focused news agencies and aggregators; regional, national and local government and intergovernmental organisations whose domain covers nuclear issues; NGOs, academic sources and blogs providing analyses on safeguards-relevant topics; general news sources and aggregators; technical publications on the nuclear fuel cycle, etc. A classification of the sources monitored by NSMM is given in Table 1 and a snapshot of the NSMM NewsBrief page is given fin Figure 3.

Some technical challenges still needed to be resolved as the nuclear websites differ from the general news media typically targeted by EMM in several aspects: i) the nuclear websites often publish unique information, thus a high reliability in the detection of new articles is required, ii) the nuclear websites are more static, i.e. the frequency of new articles/reports is much lower and iii) some of the nuclear sites require authentication.

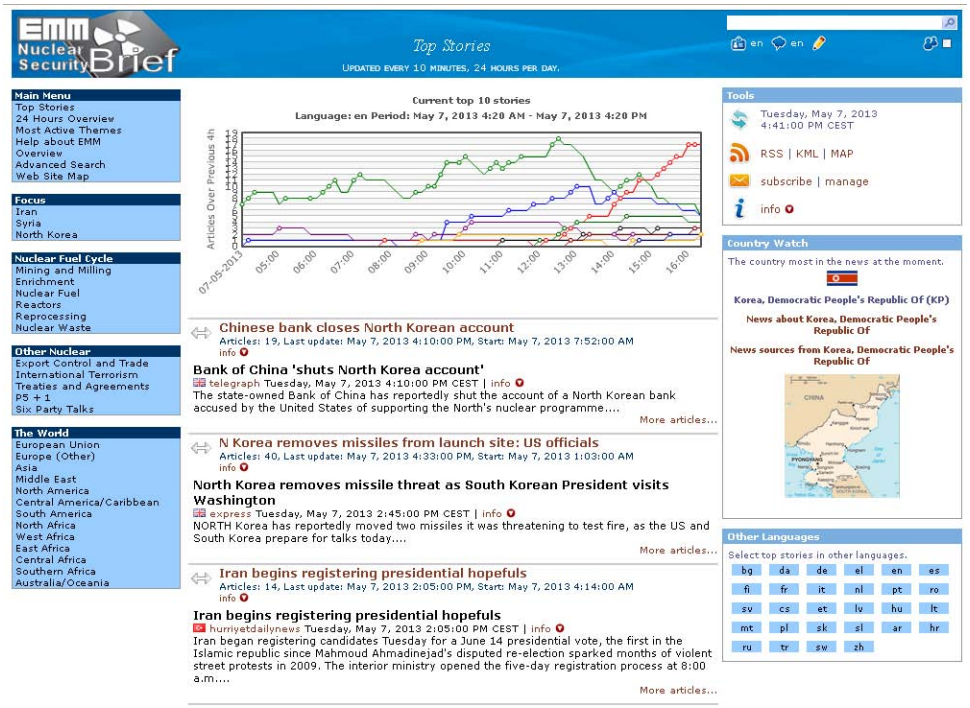


**Figure 3:** Snapshot of the NSMM NewsBrief page. Structure and functionality of the page correspond to the generic EMM NewsBrief page; however, the information content is focused on nuclear security and safeguards.

**Table 1:** Classification of sources monitored by NSMM

| Category Name | Content | Frequency |
|---|---|---|
| General News and Aggregators | Directly-accessed news sources, news aggregators, and "fee-based" comprehensive news archive with collection of newspapers, periodicals, and news wires, filtered by user defined keywords. | Very High |
| Nuclear News Aggregators | Articles from news agencies and news aggregators that customarily or primarily report on issues related to nuclear industry and safeguards. | High |
| NGO & Academic | Non-governmental organization or university providing detailed reports and added value assessments concerning State's nuclear programmes and activities, and general nuclear nonproliferation issues. | Medium |
| Blogs | Interactive websites with commentaries on nuclear issues. | Medium |
| Government & Intergovernmental | Information from relevant intergovernmental organizations and competent authorities at national level are a unique source of authoritative information on nuclear safeguards and nuclear industry issues. | Low / Medium |
| Nuclear Industry | Information on companies including location(s), products, capabilities, activities, number of employees, main customers, exports of nuclear related items. | Low / Medium |

Considerable effort was made to configure and test the monitoring of the nuclear websites: for each site, the relevant RSS feeds and/or HTML pages were selected and inserted into an EMM configuration file. The monitoring results are validated to ensure that all relevant articles are collected successfully. The maintenance of the nuclear sources is a continuous task: some of the source definitions have been optimized or updated following changes on the site or the source URL and additional sources have been added to the system on IAEA's request.

## 4.2 Interoperability with Existing IAEA Back-end IT Systems

It is important that all information which is selected during the daily review within the EMM/NewsDesk operating environment are permanently transferred to the IAEA back-end systems. The efficient archiving of information selected in NewsDesk into a backend system is essential for compatibility with IAEA workflows. As NewsDesk does not directly provide this functionality, the project aims to identify possible mechanisms for transferring the EMM/NewsDesk output (i.e. the list of selected articles including the meta-information) to the IAEA repository.

## 4.3 Testing and Deployment

### 4.3.1 Pre-launch Coverage Testing

In order to evaluate NSMM/NewsDesk performance with respect to the nuclear sources, several trials were carried out by simultaneous manual monitoring of the websites and review of articles processed in NewsDesk. Search results were then compared in number and relevance to identify the coverage of retrieved articles and required resources. The trials confirmed that the articles collected with the existing tools and workflows, could also be captured with NSMM/NewsDesk and that at a minimum the coverage provided by NewsDesk adequately replicates the coverage from the previous monitoring method.

### 4.3.2 Phase 1 Launch

After the coverage test results were found to be satisfactory, the IAEA carried out a number of usability tests to assess the efficiency of the new process in comparison with the previous method. Following weeks of intensive usability testing, the NewsDesk application was deployed in Phase 1--utilized for information review and collection. During Phase 1, web sources previously monitored manually or via a website tracker were consolidated into one source stream in the NewsDesk. Certain sources from this set remain outside the NewsDesk and continue to be reviewed manually. The consolidation affected nearly two-thirds of the search effort involved in daily monitoring, and was found to raise the overall efficiency of the GMT.

From 11-29 March 2013 SGIM GMT conducted a usability survey of NewsDesk that documented total articles reviewed, total number of relevant articles selected for review by the SGIM Open Source Highlights editor, and the total amount of search time. In this three week period, the GMT reviewed a total of 5800 documents using NewsDesk, and passed on 236 (roughly 5%) of these to the OSHL editorial team for further review.

Prior to the Phase 1 introduction of the NewsDesk application into daily operations, GMT staff members conducted searches from a set of websites and information repositories, with a collective effort of approximately three staff-hours daily. Consolidating hundreds of sources of varying structure and style into one format in the NewsDesk application that also permits a degree of de-duplication, has streamlined the daily review, reducing the effort to two-staff hours per day. The survey also found that the average length of time required to conduct the review with the help of NewsDesk was approximately one hour[1]. This represents approximately a 33% time savings in daily search and collection effort, with no loss in coverage or quality of results (see Table 2).

**Table 2:** Overall daily effort for GMT and OSHL (in staff-hours)

| | Collection[2] | Editor review and OSHL Production | Back-end processing | Total daily time requirements |
|---|---|---|---|---|
| Pre-NewsDesk launch | 3 hours | 5-7 hours | 1 hour | 9-11 hours |
| NewsDesk Phase 1 | 2 hours | 5-7 hours | 1 hour | 8-10 hours |

## 5. Future Developments

After the successful introduction of NSMM/NewsDesk into IAEA's Open Source monitoring workflow, IAEA's initial focus is to gain operational experience with the system and to evaluate different possibilities to fully benefit from the NSMM/NewsDesk capabilities. The options that are currently under consideration are listed hereafter.

### 5.1 NewsDesk for Article Collection and OSHL Production

In the next phases it is foreseen that all sources currently monitored manually should be included in the NewsDesk application, such as news aggregators, internal sources, and subscription-based websites.

In subsequent phases, the project anticipates the use of NewsDesk for the production of Open Source Highlights newsletter, which will be predicated on further integration of the NewsDesk application with SGIM workflow and infrastructure. A critical element of the process is the ability to automate the back-end processing of the newsletter information, which is expected to result in additional savings of one to two staff-hours per day.

### 5.2 NSMM for Real-time Information Awareness

Under the current scope of the project, NSMM is intended as an information source for SGIM Open Source Highlights production. A further step could be the broader distribution and use of the NSMM NewsBrief page within IAEA Safeguards. The nuclear-specific categories which automatically filter the incoming information could be further refined and thus allow the users to access the articles according to areas of interest, such as a geographic region, the different stages of the nuclear fuel cycle, or event types such as illicit trafficking of nuclear material, export control violations or dual-use equipment transfers.

The NSMM could allow an IAEA safeguards inspector or analyst involved in State evaluation to set up filters for their assigned States or facilities to ensure access to relevant open source information on a near real-time continuous basis. This passive continuous monitoring would free up (human) resources that could be re-allocated for tasks requiring more in-depth active searching. A schematic overview of a potential future setup based on NSMM and NewsDesk is illustrated in Figure 4. The user base could also be enlarged beyond SGIM; depending on the selected deployment architecture (e.g. if deployed on JRC servers), the system might even be opened as a public service to the general nuclear security community.

[1] This includes the manual review of select news aggregators, internal websites and subscription-based websites.

[2] Includes total collection time for all three source streams (see Section 2 above)

### 5.3. NSMM/NewsDesk for Country-Specific Monitoring

As well as the daily news monitoring and OSHL production, SGIM analysts also carry out country-specific monitoring and searches, which typically involves monitoring additional national sources and generates more state-specific information. Following the positive experience in the general daily monitoring, SGIM analysts are considering using NSMM/NewsDesk for country-specific monitoring. For initial trials, country-specific 'targets' i.e. separate containers where the collected information is located in NewsDesk, will be setup. State-specific daily monitoring using NSMM/NewsDesk has the potential to enhance the effectiveness and efficiency of the continuous monitoring and evaluation of states.

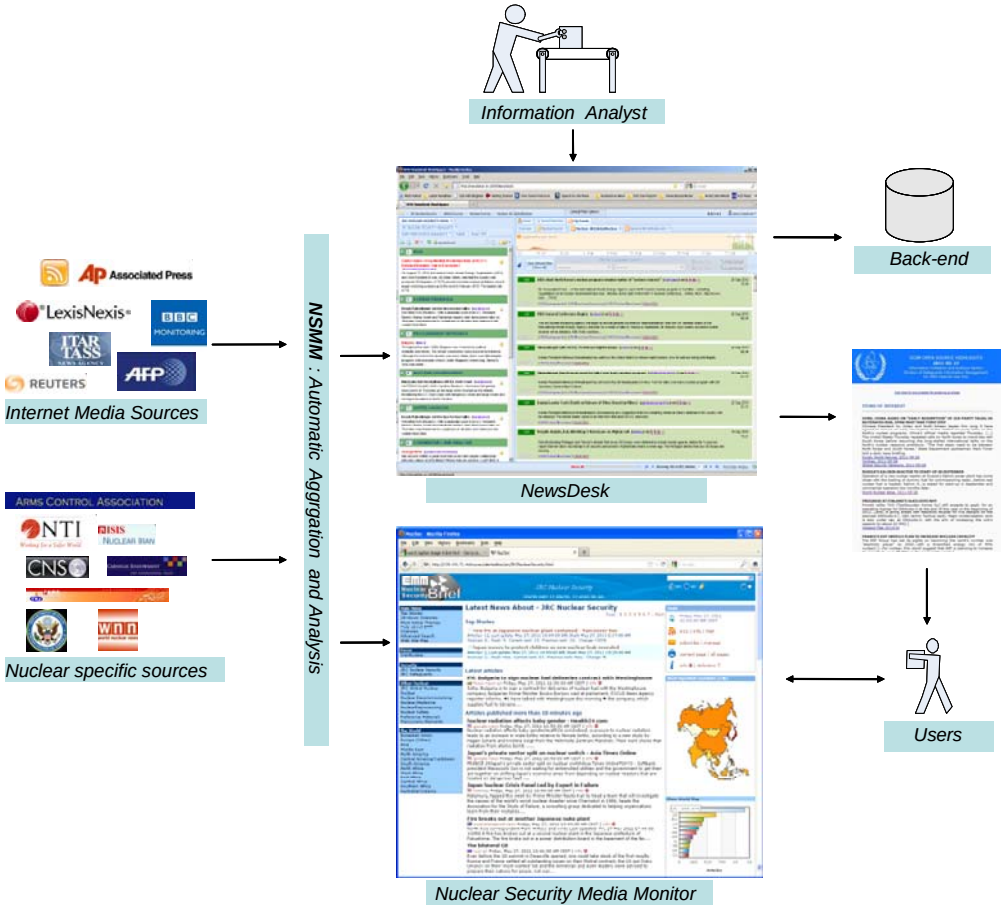


**Figure 4:** Schematic overview of information flow in a possible future setup for open source information collection and analysis based on NSMM and NewsDesk.

## 6. Conclusions

The IAEA possesses robust open source capabilities. However, the growing volume of information and variety in the number of available sources has made it imperative for the IAEA to continue to increase the effectiveness and efficiency of its collection, analysis, and dissemination processes. A clear priority for the Department of Safeguards is to streamline the acquisition and analysis of open source information and move away from the current labour intensive information collection/newsletter production process to a more efficient system.

NSMM/NewsDesk is a powerful tool developed by JRC for automatically aggregating and analysing open source information from the internet focusing on the nuclear security and safeguards domain. After extensive testing showed that utilizing NSMM/NewsDesk improves the efficiency of IAEA's open source collection process, the IAEA has started using the system in its daily workflow. Currently, the

IAEA is gaining operational experience with the system and intends to further integrate NSMM/NewsDesk into IAEA's workflow in order to fully exploit the system capabilities.

As safeguards implementation evolves to make better use of all safeguards relevant information available to the IAEA new effective and efficient approaches for open source information collection and analysis are required [10]. The utilisation of tools such as NSMM/NewsDesk together is one example of how new technologies and tools can support this process.

## Acknowledgements

The work here presented has been carried out within the European Commission (EC) Support Programme to the IAEA.

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# A new Facility for NDA-Safeguards Training

**Ludwig Holzleitner*, Jean Galy, Klaus Mayer,
Indra Krevica, Jozsef Zsigrai, Magdalena Toma**

Institute for Transuranium Elements (ITU)
Joint Research Centre, European Commission,
Hermann-von-Helmholtz-Platz 1,
D-76344 Eggenstein-Leopoldshafen, Germany

***Abstract:***

*On an initiative of the Joint Research Centre a European Nuclear Safety and Security School (EN3S) is being established. This School shall offer training on specialization topics related to the nuclear fuel cycle, one of it is "Nuclear Security, Safeguards and Forensics". For this reason the EUSECTRA project has been carried out in collaboration with DG-HOME at ITU both at its locations Karlsruhe and Ispra. Its task is to build up training facilities and to offer training concerning nuclear security, safeguards and forensics to a wide auditorium, i.e. radioprotection and border control officers, police officers and safeguards inspectors from IAEA and EURATOM. In this framework also a measurement-facility for safeguards-training concerning non-destructive analysis (NDA) is built, located in ITU at Karlsruhe.*

*In this paper the status of a new NDA training facility for safeguards, is presented. It is housed in an around 100 $m^2$ large steel container in one of the experimental halls of ITU in Karlsruhe. As in this container only encapsulated radioactive material shall be used, it establishes a contamination free controlled area. Similar safeguards training as in the long existing facility PERLA at Ispra will be provided. Inspectors from EURATOM and IAEA shall be trained there in active and passive neutron measurements, as well as in Uranium and Plutonium isotopic composition measurements using Gamma spectrometric methods and related topics.*

**Keywords:** Nuclear safeguards, nuclear security, non-proliferation, training, non-destructive analysis

## 1. Introduction

On 16 December 2011 the European Commission – Joint Research Centre launched the European Nuclear Safety and Security School (EN3S). It is an initiative to reinforce the potential of the JRC's expertise and unique facilities in graduate and post-graduate education and training, in close collaboration with academic and other educational organisations. Its educational tracks will offer specialization key areas related to the nuclear fuel cycle, among it "Nuclear Security, Safeguards and Forensics".

On increasing demands from EU member states authorities the EU CBRN action plan had been approved. Consequently the Joint Research Centre had been tasked by DG-HOME to establish an European Security Training Centre (EUSECTRA) in order to provide training concerning nuclear security and related matters to a wide auditorium, i.e. radioprotection and border control officers, police officers and safeguards inspectors from IAEA and EURATOM.

JRC has already several decades of experience in measuring and controlling nuclear material through its involvement in the safeguards area as well as long standing expertise in nuclear

* Corresponding author, E-mail: ludwig.holzleitner@ec.europa.eu

security: For many years the JRC carried out training courses in nuclear security, both at Ispra and Karlsruhe, focusing on the response to illicit incidences involving nuclear material. Training courses safeguards measurement techniques have been delivered both to IAEA and EURATOM inspectors during the last decades. On 12th of October 2012 the 20 years anniversary of nuclear forensics at JRC in Karlsruhe had been celebrated. EUSECTRA will thus profit from the JRC's in-house experience in these 3 areas of security, safeguards and forensics, which are made available.

Since the measurement techniques used for NDA in safeguards and in security are common, EUSECTRA will benefit from strong synergies between these areas, through its use of:

- similar measurement techniques (gamma spectroscopy, neutron measurement, …)
- the same trainers;
- partially the same instruments;
- the same facilities and rooms;
- Partially the same nuclear sources for training.

## 2. The EUSECTRA training facilities

On 18th of April 2013 the new EUSECTRA training facility at Karlsruhe was inaugurated, comprising of an outside training area for border control dedicated to security training and an inside training area for both security and safeguards training. The inside training area comprises a caisson of two floors, roughly 100 m2 each, and a briefing room. The ground floor is dedicated to safeguards training, the upper floor to security training whereas the briefing room is used for both.


**Figure 1:** Inauguration of EUSECTRA 18th April 2013

## 3. Security training facilities

Following the experience of nuclear security training in Ispra, the security training facility comprises an outside training area with currently 3 radiation portal monitors (RPMs) together with a control room. A fourth RPM is planed. The RPMs are located on an elliptic street, where vehicles to be controlled for training circulate. The following RPMs are available for training:

- Thermo Fisher Scientific ARIS-512: scintillation NaI spectroscopic gamma detectors together with 3He neutron detectors.

- TSA systems VM250: organic plastic scintillator detectors and 3He neutron detector tubes for neutrons.
- ASPECT YANTAR-1A: scintillation detectors and He-3 neutron detectors.
- Another spectroscopic RPM is to be installed: It shall comprise spectroscopic gamma detectors and neutron detectors.

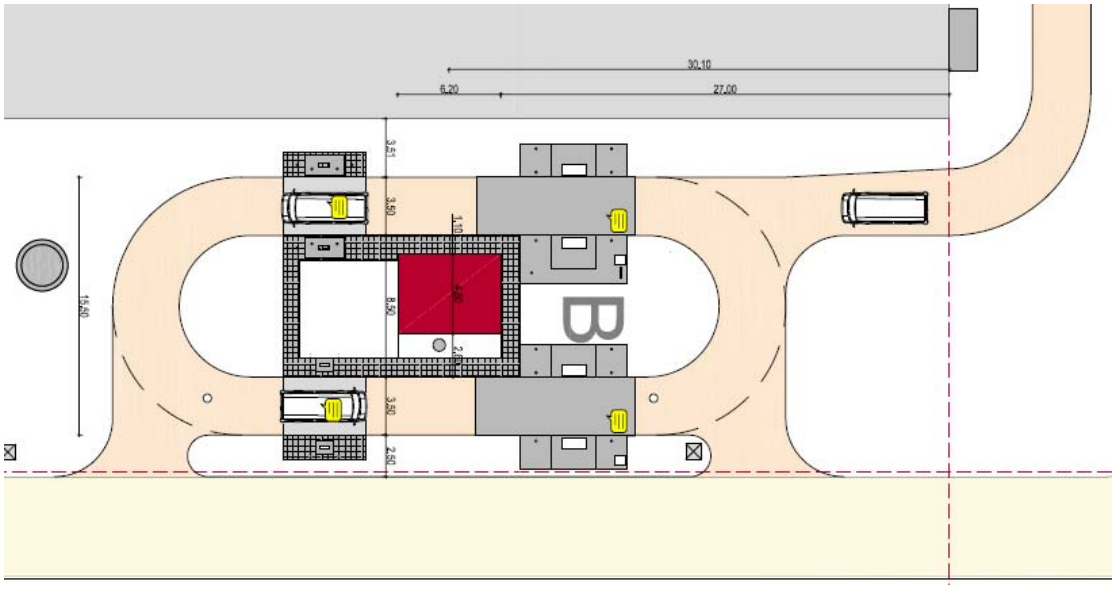


**Figure 2:** Outline of outside security training area


**Figure 3:** Outside security training area with portal monitor and container building

The RPMs are controlled from the control room, located in a container at the centre of the training area, directly between all the RPMs and the elliptic street, with windows to all sides such that all of the RPMs can be properly watched. For each RPM there is one computer

located in this container for its control. In a later stage it is planed to move the control room to another fixed building, which still has to be established.

The inside security training facility is located within the controlled area of ITU-Karlsruhe. It comprises the upper floor of a steel container (so called "caisson") within the controlled area and a briefing room. The briefing room is foreseen for theoretical lectures and will be equipped with computer work places, overhead beamer, etc.. The upper floor of the caisson contains the measurement equipment: one pedestrian radiation portal monitor and a x-ray scanner (as used at airports), several electrically cooled portable HPGe detectors with built-in multi channel analysers and isotope recognition software (ORTEC Detective), hand-held NaI detectors (identiFINDER), radiation pagers both neutron and Gamma sensitive, radiation detection backpacks, etc.. Care was taken to have several types of equipment available in order to offer the most possible variety of instruments to trainees.

**Figure 4:** security training instruments at the inauguration

The target audience for the security training courses are border control officers, radioprotection officers, special police officers (e.g. for forensics, …) national experts and decision makers, etc.. Training courses are already ongoing, the most common courses are:

- Detection at borders or nodal points;
- Train the trainers;
- Mobile expert support team;
- Response plans;
- Nuclear forensics;
- Radiological crime scene; …

A similar installation also comprising RPMs is available since a few years in Ispra. Courses are given in parallel both in Ispra and in Karlsruhe in order to cope with the high demand currently occurring in this field!

## 4. NDA safeguards training facility

In contrast to security training for safeguards training the target audience is very specific: EURATOM and IAEA inspectors. During regular inspections the commission is obligated to measure the declared fissile material in nuclear installations. One important technique for quickly doing so is by non-destructive analysis (NDA), measuring the radiation arising from such materials. Hence there is a strong overlap in measurement techniques with security field. In safeguards one refers to measurements of fissile materials: uranium, plutonium and thorium. In contrast to security measurements, where the focus is on detection of nuclear material, for NDA-safeguards measurements the demands on precision and accuracy are much higher: The fissile material needs to be determined and quantified within tight limits, so called international target values. This constitutes a key challenge of NDA-safeguards measurements.

The two most frequently used NDA techniques in safeguards are gamma ray spectroscopy and neutron measurement techniques. Gamma spectroscopy gives an isotopic composition of the measured material, but for conditions as they are usually met during in-field inspections no indication about the amount of fissile material. Neutron multiplicity counting provides with the equivalent fissile mass of the material measured. Therefore the main courses to be given in the framework of NDA safeguards training, planed to start second half of 2013, are:

- Uranium enrichment determination by Gamma spectroscopy: NaI or CdZnTe are used to make use of the higher efficiency of these detector types.
- Determination of Plutonium isotopic composition by Gamma spectroscopy: In this case HPGe detectors, either electrically cooled or cooled with liquid nitrogen need to be used since the demands on resolution are high. The isotopic composition is then calculated by the specialized software MGA.
- Passive neutron coincidence counting and determination of fissile mass (Pu240eff.): Here neutrons are moderated down to thermal energy levels, where they can be detected with $^{3}$He filled detector tubes. Through a reaction of the $^{3}$He with the neutron an electrical pulse is produced by the high voltage applied in these tubes. This pulse is recorded by an external electronics, where the number of pulses coming after a trigger pulse gives an indication about the material measured.
- Active neutron coincidence counting: This works similar to the passive neutron coincidence. Additionally the measured material is irradiated with random neutrons: During fission events induced by these random neutrons, further multiples of neutrons are released giving further indications about the measured fissile material.
- Advanced RADAR/CROSP training: This is training on a special software package: "Remote Acquisition of Data And Review". The course participants will learn to set up and configure this software in order to both perform measurements of fissile material and evaluate it remotely, form another location than where the measurement is taking place.

Of course further courses within the NDA-safeguards area could be delivered. However, feasibility, course modalities and focus needs to be discussed and agreed with potential customers individually for each of these courses:

- Calorimetric measurements? There is some expertise in house; however the need for inspection units needs to be clarified.
- NDA refresher course: This course is to refresh knowledge in the NDA-safeguards techniques of Gamma spectroscopy for uranium and plutonium isotopic composition and in active and passive neutron measurements as described above. Therefore all courses mentioned need already to be fully operational.
- Physical inventory verification (PIV): In addition to the courses mentioned before the course participants need to learn sampling techniques and how to conclude detection probabilities from these samplings and measurement uncertainties. This shall be learned during practical exercises and measurements close to realistic scenarios.
- Material unaccounted for (MUF) evaluation / statistical analysis: This is similar to the course about MUF, but with the focus the theoretical foundations of statistical theory, error propagation form measurement uncertainties, confidence level calculations and the detection probability theory of the respective sampling techniques used.

It is planed to run courses about NDA-safeguards training both in Karlsruhe and at the long standing laboratory PERLA in Ispra. The only exception is the active neutron coincidence counting course, which will only be given in Karlsruhe since the mock-up fuel element necessary for this course is located there. This will hold in future also for the instruments concerning active neutron coincidence counting.

The premises for NDA-safeguards training at Karlsruhe are the ground floor of the caisson and the briefing room. As with security training, the briefing room (which still needs to be equipped) is foreseen for theoretical lessons and pure computer training.

**Figure 5:** Caisson ground floor for NDA safeguards training

**Figure 6:** Briefing room for both security and NDA safeguards training

In the ground floor of this caisson 5 workplaces for NDA safeguards training had been established, allowing simultaneous training of maximum 10 inspectors. In contrast to security training, where a variety of instruments is desirable, in safeguards we need to use the same instruments as the respective inspectors in order to maximize the training effect. A great deal of instruments for NDA safeguards training had been acquired:
Concerning Gamma spectroscopy:

- One electrically cooled HPGe detector (CP5 from Canberra)
- One MCA 527
- 4 MCA166 (loan from DG-ENER)
- 4 liquid nitrogen cooled HPGe detectors (loan from DG-ENER)
- 2 NaI and 2 CdZnTe detectors (loan from DG-ENER)

Concerning neutron multiplicity counting:

- One Plutonium scrap multiplicity counter (PSMC, loan from DG-ENER)
- One HLNCC (loan from DG-ENER)

- 1 PTR-32HV list mode pulse train recorder

This equipment described above allows being fully operational and self-sufficient concerning gamma training courses. Some equipment, mainly for active but also passive neutron multiplicity counting has been committed by DG-ENER. It is expected to be received within the next few month in order to become operational for neutron coincidence counting as well:

- 2 more HLNCCs
- 4 JSR12 (or JSR14)
- 1 neutron coincidence collar (NCC, with associated neutron sources)
- 1 active well coincidence counter (AWCC, with associated neutron sources)

A small computer network with a separate domain was set up in order to train inspectors in remote data acquisition: The trainees have to set up the measurement equipment in the caisson, but control it remotely from the briefing room to draw conclusions using specialised software called RADAR/CRISP.

The following samples of fissile material, partially received as a loan from DG-ENER, partially transferred from Ispra, are present for training:

- 3 sets of CBNM Uranium standards with different enrichment
- 1 set of CBNM Pu standard with different isotopic
- 4 PIDIE standards with different isotopic
- Pu pellets N1 to N10, N20, N21, N22, N30 and N40 with different masses
- 2 Uranium metal standards with different enrichment respectively
- 2 Mock-up fuel assemblies (one for BWR, one for PWR) with some spare material. A small crane is available in order to perform measurements at the fuel elements, which for this reason can be lifted out from their carrying-frames.


**Figure 6:** Samples to be measured for NDA safeguards training

## 6. Conclusion - summary

On 18th of April 2013 inauguration of the European Security Training Centre (EUSECTRA) took place at ITU in Karlsruhe. It had been established within the framework of the European Nuclear Safety and Security School (EN3S).

It is composed of an outside training area for security training, equipped with portal radiation monitors (RPMs), a caisson inside the controlled area where the upper floor is dedicated to security training, the ground floor for safeguards training. A briefing room is to be used for both theoretical lessons in both security and safeguards training. The facilities are readily equipped with instruments for gamma spectroscopy and neutron multiplicity counting, however some for neutrons measurement are still to arrive. Samples of radioactive material to be measured are present in sufficient numbers.

Its training activities are on the one hand security training for a wide auditorium such as border control and police officers, decision makers, radioprotection officers, etc. on the other hand rather specialised NDA safeguards training for inspectors of EURATOM and IAEA. Since techniques used for both applications are similar, there are various synergies between these training areas: Use of the same rooms, facilities, instruments, trainers and training samples can be made. While security training is already operational and courses are presently given, safeguards-training is planed to start in autumn 2013. Training courses in safeguards and security will be given in parallel both at Karlsruhe and in already existing, similar facilities in Ispra. The only exception is a course about active neutron coincidence counting for safeguards, which will be solely held at Karlsruhe.

## 7. Acknowledgements

Here we want to take the opportunity to thank the numerous persons who contributed to establishing EUSECTRA and therefore made the success of the inauguration possible. As there are various external companies, in-house craftsmen, authorities, scientists, administrations and many more. Special acknowledgements to DG-ENER, DG-HOME and DG-DEVCO for collaboration, strategic development and support.

# Tracking of fissile material by means of coincident neutron detection - Fission Meter vs. Slab Counter

**T. Köble[1], W. Berky[1], H. Friedrich[1], M.Risse[1], W. Rosenstock[1], O. Schumann[1], R. Berndt[2]**

[1]Fraunhofer Institut Naturwissenschaftlich-Technische Trendanalysen (INT)
Appelsgarten 2, D-53864 Euskirchen, Germany,
[2]Joint Research Centre, 21020 Ispra, Italy
E-mail: theo.koeble@int.fraunhofer.de

***Abstract:***

*In the context of fissile material detection, discrimination between changes of the natural neutron background, industrial material, and a real neutron source, indicating the presence of nuclear material, is tremendously important. The negligence of such material may lead to drastic if not fatal consequences to the general public in case the material in question is used in explosive devices such as Improvised Nuclear Devices (IND). To achieve the discrimination required, measurements of coincident neutrons are feasible because only fissile material emits coincident neutrons. Two neutron detection systems, the Fission Meter by Ametek/Ortec and the Slab Counter by Canberra, were tested concerning their quality of detecting nuclear material as mentioned above, using primarily a variety of Plutonium sources with different isotopic compositions provided by the Institute of Transuranium Elements (ITU) of the Joint Research Centre at Ispra, Italy, where the measurements were performed. We also examined the influence of shielding effects of the materials which would possibly be part of an IND. The results of these measurements and an assessment of the reliability of the two detection systems concerning fissile material verification will be presented.*

**Keywords:** neutron measurements, coincident neutrons, fissile material detection

## 1. Introduction

The current threat of the use of nuclear material during terrorist attacks, e.g. by means of improvised nuclear devices, is a widely discussed matter nowadays. Such devices may have severe consequences to the general public. In any such case it is vital to investigate which type of nuclear material could be involved, therefore sophisticated measurement techniques are required. Whereas gamma measuring devices are more commonly used, the utilization of neutron measurement techniques may be very valuable in some cases as the nuclear device could be surrounded by shielding material which gamma radiation cannot pass. Neutrons, on the other hand, may still be detectable. Additionally, the neutron background is usually low.

The neutron measuring component of common hand-held radiation measuring devices is not very suitable for such measurements because its detection volume is generally too small. Large volume neutron detection devices are required to distinguish between nuclear and industrial material by means of detecting the presence of coincident neutrons.

## 2. Neutron detection systems

Two neutron detection systems suitable for measurements of coincident neutrons were compared to verify their ability to reveal the presence of nuclear material. These systems are the Fission Meter by Ametek/ORTEC and the Slab Counter by Canberra. Both systems are equipped with $^{3}$He-tubes for neutron detection and polyethylene coating for moderation of fission neutrons as $^{3}$He is best suited for detecting neutrons in the thermal energy range. The most important physical parameters of the

devices are listed in table 1. Their most significant difference in appearance is the fact that the Fission Meter comprises two panels connected by hinges which allows for flexible arrays of the panels with different angles relative to each other (see figure 1 on the left). The Slab Counter, respectively, only consists of a single panel (see figure 1 on the right). Various Slab Counters can be positioned around a source to match different geometries and to achieve an optimal efficiency. For the comparison in this experiment, two Slab Counters were used in order to match the measuring options with the Fission Meter.

| Type | Fission Meter | Slab Counter |
|---|---|---|
| Manufacturer | Ametek/ORTEC | Canberra |
| Gas characteristics | $^3$He ($7.6 \cdot 10^5$ Pa) | $^3$He ($4.2 \cdot 10^5$ Pa, $5 \cdot 10^4$ g/cm$^3$) |
| Diameter per tube | 2.54 cm | 2.54 cm |
| Length per tube | 48.26 cm | 33 cm |
| Number of tubes | 15 per panel<br>=> 30 per device | 6 tubes per counter |
| Active area | ~1800 cm$^2$ (15 tubes, 0° geometry) | ~500 cm$^2$ |
| Moderator | Polyethylene, on one side (minimum 2.54 cm) | Polyethylene, enclosing the tubes (minimum 1.7 cm) |
| Weight | 26 kg | 11.5 kg |

**Table 1**: Comparison of physical parameters of neutron detection systems.

The Fission Meter system can be operated in three different modes [1]:

- Mobile search mode
- Static search mode
- Characterization data collect mode

In the mobile search mode temporal changes of the neutron count rate are displayed. This mode is therefore not suitable for static measurement set-ups as described in this paper. In the static search mode, discrimination between the total count rate and the non-cosmic count rate is shown. The cosmic part of the count rate refers to non-correlated neutrons caused either by the natural neutron background which creates a yield when entering the detection volume or industrial material emitting non-coincident neutrons. The non-cosmic part consists of neutrons emitted by sources generating coincident neutrons by spontaneous fission. So measurements performed in this mode give a hint if the count rate values are created by fissile material, industrial material, or just the natural background.

Based on the measured count rate data, the Fission Meter generates multiplicity plots in the characterization data collect mode, displaying the distribution of the counts over the multiplicity numbers. In case these plots show a Poisson distribution, only randomly emitted neutrons are present. A distribution of a neutron source representing industrial material should be similar to one caused by the natural background. In contrast, fissile material such as Plutonium emits coincident neutrons, so the distribution of the multiplicity numbers does not equal that of a purely random emission. Thus, by detecting coincident neutrons, the detection systems can verify the presence of fissile material. For further details see, e.g., [2].

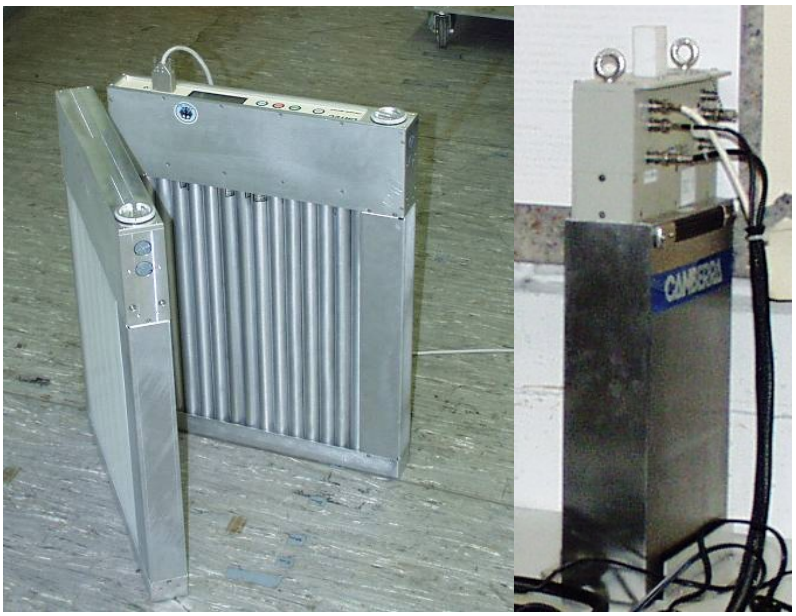

**Figure 1**: Detection panels of the Fission Meter (left) and the Slab Counter (right).

The Slab Counters are also capable of measuring coincident neutrons by means of multiplicities. They measure the Singles, Doubles and Triples rates as defined by Ensslin et al [3]. From these rates it is possible to determine the fissionable mass, multiplication and alpha if the efficiency of the detector set-up is known. The efficiency in general is not known precisely for a mobile set-up intended for the in-situ measurement of unknown radioactive material as described here. The efficiency of various Slab Counter set-ups with 4, 6 and 8 slab counters had been measured with different $^{252}$Cf sources at Fraunhofer-INT previously [4]. For the measurements here where only two slab counters were used two $^{252}$Cf sources of the ISPRA facility were used for efficiency calibration.

The efficiency can be determined from the Singles rates of the $^{252}$Cf sources assuming that the alpha value is 0 and the multiplication is 1.

## 3. Measurement set-up

The measurements were carried out at the Institute for Transuranium Elements (ITU) at the Joint Research Center (JRC) in Ispra, Italy. The ITU provided lab space and, most importantly, the neutron sources used for this study. The neutron measurement devices belong to the Fraunhofer-INT.

The arrangement of the detectors or its panels affects, among other factors, their efficiency. According to previous measurement results, the Fission Meter's maximum efficiency is achieved when the panels are placed at a relative angle of 45° to each other with the moderation material on the outside, the source being wedged between the panels (see figure 2). If the source is placed in the center of the geometry the corresponding efficiency value is approximately 6 % [5]. In the set-up used here the source was placed outside the center of the geometry at 24 cm distance to the center of the panels. The efficiency determined by the measurement of the two $^{252}$Cf sources was 1.42 +/- 0.045 %. With the Slab Counters, several geometries were tested. The measurements were performed with the two devices facing each other with the source placed in between (see figure 2). The distance between the front faces of the panels was 23 cm and the corresponding efficiency was determined as 4.85 +/- 0.11 %. It was tried to position the sources in the vertical and horizontal middle position of the active detector surface as good as possible.

**Figure 2**: Measurement set-up of the Fission Meter (left) and of the Slab Counters (right).The distance of the source on the tripod to the center of one of the Fission Meter pannels was 24 cm. The distance between the front surfaces of the Slabs was 23 cm

Figure 3 shows the electronic components and laptops for the two systems. The Fission Meter's data taking device is a ruggedized pocket PC connected by a serial cable. The Slab Counters are controlled by a JSR14 which comprises the high voltage unit and a shift register.

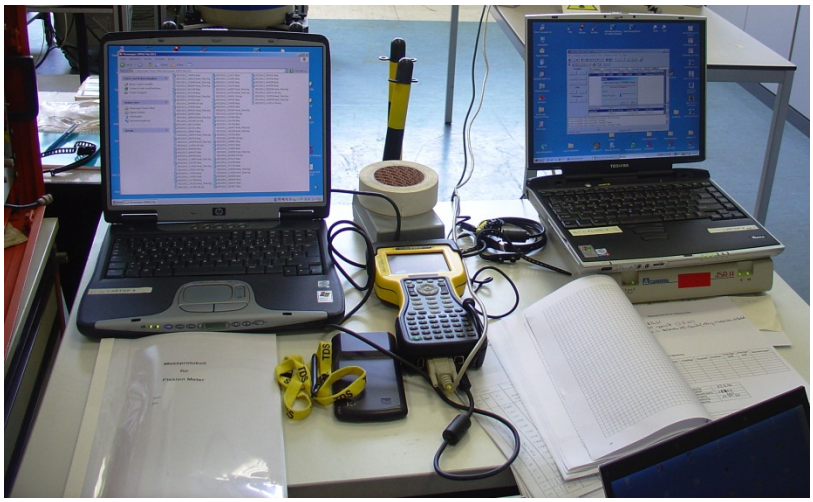

**Figure 3**: The Fission Meter data collection device (in the center of the picture), the Slab Counter high voltage supply and data collection unit (right, beneath the notebook) and notebooks.

In order to test the systems' ability to detect nuclear material by coincident neutrons, two $^{252}$Cf sources were used for calibration measurements. As nuclear material several Pu sources with different isotopic compositions, ranging from Reactor Grade Pu (RG Pu) to Weapons Grade Pu (WG Pu) were used. The Plutonium masses and isotopic compositions of these sources are listed in table 2. In addition to the Pu sources two MOX sources were measured, but only with the Slab Counters for logistic reasons.

Measurements were also performed with additional shielding which consisted of a 1.5 cm thick layer of explosives simulat around the source.

| Source | Mass Pu [g] | Pu-239 Content [At.%] | Pu-240 Content [At.%] |
|---|---|---|---|
| Pu CBNM 61 | 6.626 | 62.7 | 25.4 |
| Pu CBNM 70 | 6.665 | 73.4 | 18.2 |
| Pu CBNM 84 | 6.690 | 84.4 | 14.2 |
| Pu CBNM 93 | 6.625 | 93.4 | 6.3 |
| PuO2 No. 1-10 | 1.98-2.0 | 85.2 | 13.2 |
| PuO2 No. 20a | 4.98 | 85.2 | 13.2 |
| PuO2 No. 21a | 9.97 | 85.2 | 13.2 |
| PuO2 No. 22a | 19.92 | 85.2 | 13.2 |
| PuO2 No. 30 | 20.57 | 69.3 | 26.3 |
| Pu Metal No. 40 | 8.45 | 89.0 | 10.2 |
| MOX ENEA-01 | 1093 | 66.3 | 28.1 |
| MOX ENEA-02 | 1093 | 66.4 | 28.4 |

**Table 2**: Overview of neutron sources investigated in this survey.

## 4. Measurement results

In the table 3 the total count rate and the non-cosmic count rate of the Fission Meter measurements for all sources is listed. We performed measurements in both the static and the characterization mode. If the time did not allow us to use both modes, we concentrated on the latter mode as it provides a more valuable output. In these cases, the non-cosmic part of the count rates could not be listed in table 3.

The count rate values of Californium and Plutonium sources measured with the Fission Meter were all significantly above the background value (see table 3) and were created almost completely of non-cosmic neutrons, indicating the presence of fissile material clearly. The addition of shielding material generally led to an increase of the count rate values which shows that the moderation material of the detection unit of the device alone is insufficient for thermalizing the incoming neutrons. Unfortunately, the background value of the count rate also comprises a non-cosmic part which could not be prevented.

| Source and Geometry | Count Rate [cps] Total (N-Cosmic) |
|---|---|
| None (Background) | 5.2 (4.6) |
| Cf-252 6001 NC, no shielding | 1758.5 (1758.5) |
| Cf-252 542 H9-716, no shielding | 4405.9 (3701) |
| PuO2 (0.2 g) with 13 % Pu-240 No. 10, no shielding | 14 |
| PuO2 (0.2 g) with 13 % Pu-240 No. 2, no shielding | 13 |
| Pu CBNM 61, no shielding | 60.32 (59.7) |
| Pu CBNM 61, explosive simulate shielding | 77.76 (76.2) |
| Pu CBNM 70, no shielding | 46.26 (44.9) |
| Pu CBNM 70, explosive simulate shielding | 59.8 (59.8) |
| PuO2 (0.2 g) with 13 % Pu-240 No. 1a, 2, 3, 4a, 5, 6, 7a, 8, 9, 10, 20a, 21a, 22a, no shielding | 171 |
| Pu CBNM 93, no shielding | 14.7 (14.3) |
| Pu CBNM 93, explosive simulate shielding | 18.04 (18) |
| Pu Metal 8.4 g No. 40, no shielding | 24.5 (24.5) |
| Pu Metal 8.4 g No. 40, explosive simulate shielding | 29.2 (29.2) |
| Pu CBNM 84, no shielding | 22.7 (22.7) |
| Pu CBNM 84, explosive simulate shielding | 28.2 (27.9) |
| PuO2 (0.2 g) with 13 % Pu-240 No. 22a, no shielding | 75 |
| Pu PM1, PM2, PM3, 13 g + 2 x 19 g, no shielding | 59 |

**Table 3**: Total count rates and non-cosmic count rate for all neutron sources measured with the Fission Meter

Figure 3 shows the comparison of two multiplicity plots created by the Fission Meter's characterization mode. The plot of the background measurement on the left differs from a purely random Poisson distribution. This may have occurred because of influences of shielded neutron sources, being in storage in the vicinity. However, the multiplicity numbers of the distribution are low. In contrast, a Pu source (figure 3 on the right) shows higher multiplicity numbers and larger differences between the measured data and the Poisson distribution. This demonstrates that fissile material was indeed detected here.

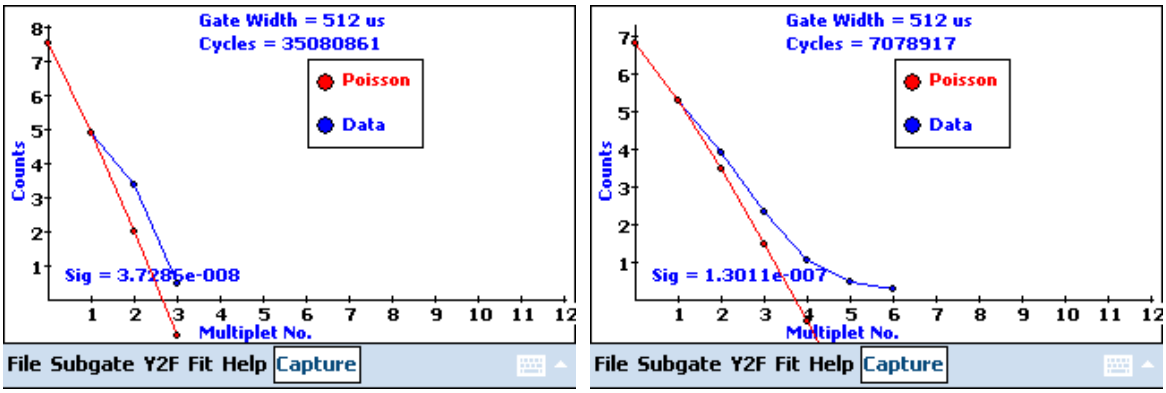


**Figure 3**: Multiplicity plots of the Fission Meter for a background measurement (left) and with the source Pu CBNM 61 (right).

Table 4 shows the count rates for the different sources measured with the Slab Counters. In general the rates for the samples with shielding are higher than the rates without shielding. This is true for both measurement systems and is caused by the moderator thickness which is too low for optimal

moderation of fission neutrons. Thus additional moderator material, which the explosives simulate used as shielding represents, enhances the count rate.

| Source and Geometry | S [cps] | ΔS [cps] | D [cps] | ΔS [cps] | T [cps] | ΔS [cps] |
|---|---|---|---|---|---|---|
| None (Background) | 1.2 | 0.06 | 0 | 0 | 0 | 0 |
| Cf-252 6001 NC, no shielding | 5955.69 | 3.16 | 292.77 | 2.53 | 7.66 | 1.48 |
| Cf-252 542 H9-716, no shielding | 15254.94 | 5.08 | 752.58 | 5.96 | 22.94 | 5.34 |
| PuO2 (0.2 g) with 13 % Pu-240 No. 10, no shielding | 22.65 | 0.11 | 0.37 | 0.02 | 0.01 | 0 |
| Pu CBNM 61, no shielding | 198.32 | 0.23 | 3.22 | 0.05 | 0.07 | 0.01 |
| Pu CBNM 61, explosive simulate shielding | 215.68 | 0.16 | 3.67 | 0.03 | 0.09 | 0.01 |
| Pu CBNM 70, no shielding | 146.68 | 0.2 | 2.23 | 0.04 | 0.04 | 0.01 |
| Pu CBNM 70, explosive simulate shielding | 159.33 | 0.21 | 2.54 | 0.04 | 0.05 | 0.01 |
| Pu CBNM 93, no shielding | 35.88 | 0.07 | 0.6 | 0.01 | 0.01 | 0 |
| Pu CBNM 93, explosive simulate shielding | 38.66 | 0.1 | 0.69 | 0.02 | 0.01 | 0 |
| Pu No. 30, no shielding | 478.42 | 0.4 | 8.72 | 0.11 | 0.18 | 0.02 |
| Pu Metal 8.4 g No. 40, no shielding | 45.01 | 0.03 | 1.26 | 0.01 | 0.02 | 0 |
| Pu Metal 8.4 g No. 40, explosive simulate shielding | 49.31 | 0.12 | 1.47 | 0.02 | 0.03 | 0 |
| Pu CBNM 84, no shielding | 69.3 | 0.14 | 1.39 | 0.02 | 0.02 | 0 |
| | | | | | | |
| Background PERLA | 40.81 | 0.03 | 0 | 0 | 0 | 0 |
| MOX ENEA-01 Perla | 3630 | 1.01 | 54.69 | 0.64 | 1.14 | 0.26 |
| MOX ENEA-02 Perla | 4453.95 | 3.86 | 71.19 | 2.65 | 0.48 | 1.2 |

**Table 4**: Single count rate (S), Double count rate (D) and Triple count rate (T) for all neutron sources measured with the Slab Counters

Due to limited measurement time the needed statistical accuracy for the Triples rate could not always be reached. The uncertainty in the Triples rate is the main factor in the uncertainty of the $^{240}$Pu effective mass. Table 5 shows the $^{240}Pu_{eff}$ values determined from the slab counter measurements.

| Source | Mass Pu [g] | $^{239}$Pu Content [At.%] | $^{240}$Pu Content [At.%] | $^{240}Pu_{eff}$ mass declared [g] | measured [g] |
|---|---|---|---|---|---|
| Pu CBNM 61 | 6.626 | 62.7 | 25.4 | 2.34 | 2.02 |
| Pu CBNM 70 | 6.665 | 73.4 | 18.2 | 1.59 | 1.45 |
| Pu CBNM 84 | 6.690 | 84.4 | 14.2 | 0.999 | 1.06 |
| Pu CBNM 93 | 6.625 | 93.4 | 6.3 | 0.423 | 0.34 |
| PuO2 No. 30 | 20.57 | 69.3 | 26.3 | 5.96 | 5.23 |
| Pu Metal No. 40 | 8.45 | 89.0 | 10.2 | 0.889 | 0.86 |
| MOX ENEA-01 | 1093 | 66.3 | 28.1 | 53.77 | 31.7 |

**Table 5**: $^{240}$Pu effective mass determined by the Slab Counter measurements compared to the value of the $^{240}$Pu effective mass calculated from the source certificates

## 5. Conclusion

It could be shown that the Fission Meter and the Slab Counters are able to detect the presence of fissionable material if an unknown object containing radioactive material is discovered. Furthermore the slab counter measurements showed that with a simple efficiency calibration the effective $^{240}$Pu mass can be estimated. This worked quite satisfactory for the available reference sources. For the Fission Meter the multiplicity distributions have to be evaluated visually on the screen of the pocket PC. Thereby fission sources can be discriminated from random neutron sources. The Fission Meter and the Slab counters are valuable devices for the determination of the presence of fissionable material in an object containing radioactive material and thus an important tool to prevent illicit trafficking of nuclear material.

## 6. Acknowledgements

The authors would like to thank the laboratory staff at JRC in Ispra, especially Mr. S. Frison, for their excellent support.

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# Design of a liquid scintillator-based prototype neutron coincidence counter for Nuclear Safeguards

**Alice Tomanin[1,2], Paolo Peerani[1], Hamid Tagziria[1], Greet Janssens-Maenhout[2], Jan Paepen[3], Anthony Lavietes[4], Romano Plenteda[4], Nicholas Mascarenhas[4], L. Marie Cronholm[4]**

[1]EC-JRC-ITU Ispra
[2]University of Ghent, Belgium
[3]EC-JRC-IRMM Geel
[4]International Atomic Energy Agency
E-mail: alice.tomanin@jrc.ec.europa.eu

**Abstract:**

A liquid scintillator-based neutron coincidence counting system designed to address a number of safeguards applications is under development by the IAEA in collaboration with the Joint Research Centre-ITU and Hybrid Instruments LTD.
Liquid scintillators are a promising technology due to their good fast-neutron detection capabilities. The characteristic fast response of scintillators is particularly beneficial for coincidence counting applications, for which a performance level higher than that associated with moderated thermal detectors might be expected. Fast neutron detection requires no thermalization process and therefore, does not incur the resulting neutron detection delays. These features reduce the length of the coincidence gate by three orders of magnitude, reducing practically at negligible values the accidental coincidence rate which dominates the uncertainty in thermal neutron detectors. Recent progress in fast electronic digitizers offers the possibility to perform on-line, real-time pulse shape discrimination (PSD) between gamma and neutron radiation detection, making this technology suitable for nuclear safeguards and security applications.
This paper will describe the experiments and Monte Carlo modeling activities engaged to design a prototype liquid scintillator-based neutron coincidence collar for fresh fuel assembly verification.
The characterization of the system response required accurate calibration measurements in order to determine the operational parameters of the liquid scintillator cell, including gain, pulse shape discrimination and energy thresholds.
Extensive Monte Carlo simulations which are essential for the understanding and characterization of the system's response were also carried out using the MCNPX-PoliMi Monte Carlo code to simulate the radiation transport within the system and to optimize the detector design. The evolution from the different detector configurations we investigated to the characteristic features of the final design will be described.

**Keywords:** non-destructive assay; neutron detection; coincidence counting; liquid scintillators; Monte Carlo modeling.

## 1. Introduction

Coincidence counting is a well-known measurement technique commonly used in nuclear safeguards for the verification of the declared quantity of special nuclear material.
The technique relies on the detection of time-correlated neutrons from either spontaneous or induced fissions occurring in a nuclear material containing fissile or fissionable material such as Uranium and Plutonium. In the presence of a well-known calibration curve, the rate of coincident neutron detection events can be directly related to the mass of such isotopes in the investigated material.
In order to detect neutrons, currently deployed coincidence counters rely on the use of $^{3}$He gas, due to its high cross section for thermal neutron captures in addition to other advantages. The recently

increased demand for $^{3}$He-based neutron detectors also for nuclear security applications, coupled with the limited production of $^{3}$He, has made this gas practically unavailable, creating the need to search for alternative neutron detection solutions.
The International Atomic Energy Agency, in collaboration with the Joint Research Centre - Institute for Transuranium Elements and Hybrid Instruments LTD is developing a liquid scintillator-based neutron coincidence counter to replace the current deployed systems for safeguards applications.
The choice of liquid scintillators as a suitable $^{3}$He alternative for this particular application was motivated by the very fast response of this detection medium. In addition, improved performance as compared to classical $^{3}$He counters is expected with regard to measurement time and related statistics. Fast neutron detection requires no thermalization process and therefore, does not incur the resulting neutron detection delays (i.e., die-away time). These features reduce the length of the coincidence gate by three orders of magnitude, reducing practically at negligible values the accidental coincidence rate which dominates the uncertainty in thermal neutron detectors. Recent progress in fast electronic digitizers offers the possibility to perform on-line, real-time pulse shape discrimination (PSD) between gamma and neutron radiation detection events, making this technology suitable for nuclear safeguards and security applications.
The characterization of the performances of the system and its design optimization were performed by means of Monte Carlo simulations. A careful validation of the simulations was performed on a small scale coincidence system (composed of two liquid scintillator cells) proving that the modeling could realistically reproduce the detector response.

## 2. Measurements set-up

The prototype system is composed of an array of EJ-309 liquid scintillator cells, with cubic geometry of 10 cm width, a multichannel real-time pulse shape discrimination (PSD) system, and a high-speed data acquisition and signal processing system to compute coincidences. The detailed design and electronics setup is discussed by Lavietes et al. [1].
A prototype liquid scintillator cell which has the same geometry and size as the one we foresee to use for the full-scale system was characterized at the JRC-ITU laboratories in Ispra, at the JRC-IRMM in Geel as well as at the PTB facilities in Braunschweig, Germany. Radionuclide gamma sources and monoenergetic neutron beams were used in order to determine the response of the detector to different radiation types and energies.
For the coincidence validation measurements, a second EJ-309 liquid scintillator of cylindrical geometry (5 inches diameter and 3 inches length) was used.

## 3. Liquid scintillator cell characterization

### 3.1 Operating Voltage

The selection of the optimal operating voltage for the detector was carried out by evaluating the detector resolution and linearity for different gamma energies. Figure 1 shows the measured relative FWHM at the Compton edge with respect to the applied voltage for three gamma energy lines.

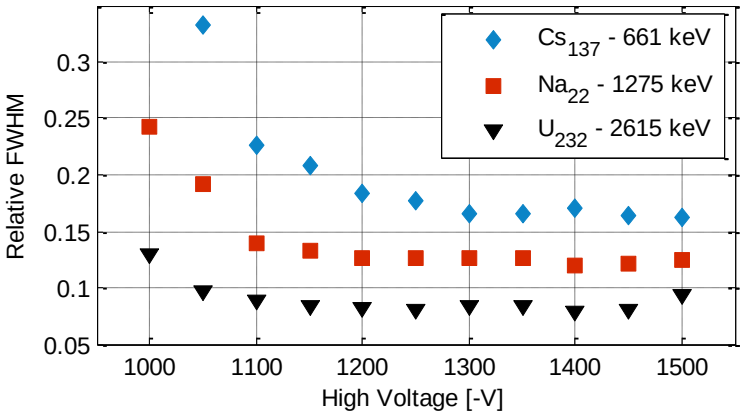


Figure 1. Relative FWHM at the Compton edge with respect to applied High Voltage for three different gamma energy lines.

As expected, the resolution increases at higher voltage as a result of the higher signal generated by the photomultiplier tube. Within a certain voltage range, the system response is fully linear, meaning that the ratio between two Compton peaks of different energies is a constant value, whereas at higher voltages it deviates. This effect appears at lower voltages for higher energy gammas, for which more photoelectrons are generated, therefore the linearity range is smaller when evaluated for higher energy gammas.
The optimal operating voltage was defined as the maximum voltage at which the system response presents full linearity in a wide energy range, measured with a $^{137}$Cs (661 keV) and an $^{241}$AmBe (4443 keV) sources, and was set to -1250 V. Further investigations also confirmed that this voltage optimizes the performances of the pulse shape discrimination as well.

### 3.2 Pulse Shape Discrimination

The pulse shape discrimination (PSD) electronics implemented in the system is based on comparing the peak amplitude to the amplitude in the decay face of the pulse, i.e. the amplitude at 16 ns after the peak. The latter is referred to as discrimination amplitude. This approach eliminates the need to integrate the charge under the pulse, which is one of the most common PSD techniques and performs much faster thus allowing real-time discrimination and real-time coincidence computation.
Figure 2 represents a typical PSD graph for a $^{252}$Cf source, where gamma pulses are represented as red points and neutron pulses as blue points. The discrimination between the two clouds of events is given by a two segment line, defined by the coordinates of the points A, B and C.

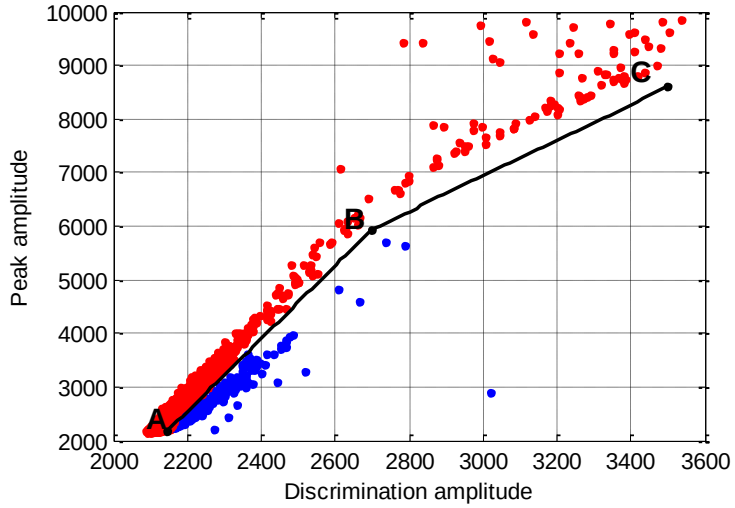

Figure 2. Typical PSD graph for a Cf252 measurements. Gamma pulses are shown in red, neutron pulses in blue. Amplitudes as measured in digitizer units.

Optimization of the PSD was obtained by varying of the intercept of the discrimination line in the discrimination amplitude axis (i.e. by variation of the x-coordinate of the A point), and analyzing the PSD response of the system in terms of neutron intrinsic efficiency and gamma rejection rate. Figure 3 represents the resulting Figure of Merit of this analysis.
The gamma rejection rate (GARR), defined as the ratio of misclassified neutrons in the presence of a pure gamma source to the totals detected pulses, can thus be written as:

$$GARR = \frac{\text{Neutrons}}{Neutrons + Gammas}$$

It is well known that the most probable misclassification of neutron and gamma events occurs in the low energy region, therefore the GARR was computed for a measurement with an $^{241}$Am source. The intrinsic efficiency values are derived by a measurement with a $^{252}$Cf source. For these measurements, the light energy threshold, which will be described in more detail in a following section, was set to 385 KeVee.
The resulting PSD settings were determined by an optimization of neutron detection efficiency and gamma rejection. For the selected setting, the neutron intrinsic efficiency is 14% and the GARR is 0.05%.

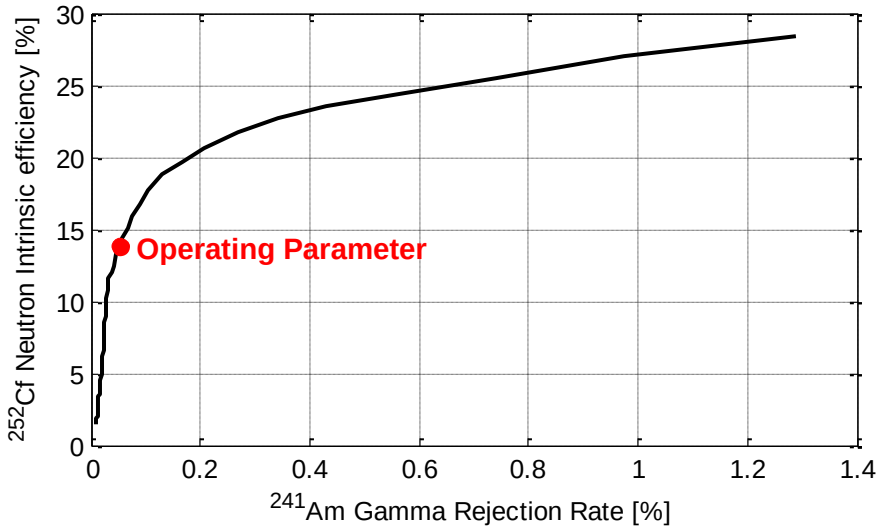


Figure 3. PSD Figure of Merit showing the GARR and the neutron intrinsic efficiency response at the variation of the Ax discrimination parameter. The red circle represents the operating parameter.

### 3.3 Validation of simulated detector response

The scintillation response of the detector, in terms of light production per incident particle and energy, is required for reliable simulation of the detector's response. Liquid scintillators typically present a linear response for gammas, whereas the light produced by protons and heavier particles varies non-linearly with energy [2]. The light output function (LOF) for recoil protons (the detection mechanism for neutrons) depends on the liquid type, the cell geometry and the cell–photomultiplier coupling. The LOF for cylindrical EJ-309 liquid scintillator of different diameters has been already reported by Pozzi et al. [3, 4], but the particular geometry of our detector requires a specific derivation of its light response.

The calibration measurements were performed at the PTB laboratories in Braunschweig, Germany, and consisted in time-of-flight measurements using the PTB cyclotron and measurements with quasi-monoenergetic neutron beams at the PTB Van Der Graff accelerator. The analysis of the data is still in progress. We report here on results obtained with a LOF obtained following a preliminary analysis only of the data. A detailed description of the determination of the LOF for this detector will be subject of a separate publication.

The results of the Monte Carlo simulations have been validated by comparing the simulated and experimental response for gamma rays from a $^{137}$Cs and $^{60}$Co source and for neutrons from a $^{252}$Cf spontaneous fission source. The results for the gamma ray measurements were used to determine the conversion factor that relates the observed amplitude to a light output in electron equivalent for the data acquisition system. Figure 4 compares the measured and simulated pulse height spectra for $^{137}$Cs and $^{60}$Co sources. The measurements are affected by a high background in the low energy region, but in the region of the Compton edges there is a good agreement between the simulated and experimental response. The simulated response is overestimated by about at least 15%. The uncertainty due to the counting statistics on the detection efficiency is about 2%.

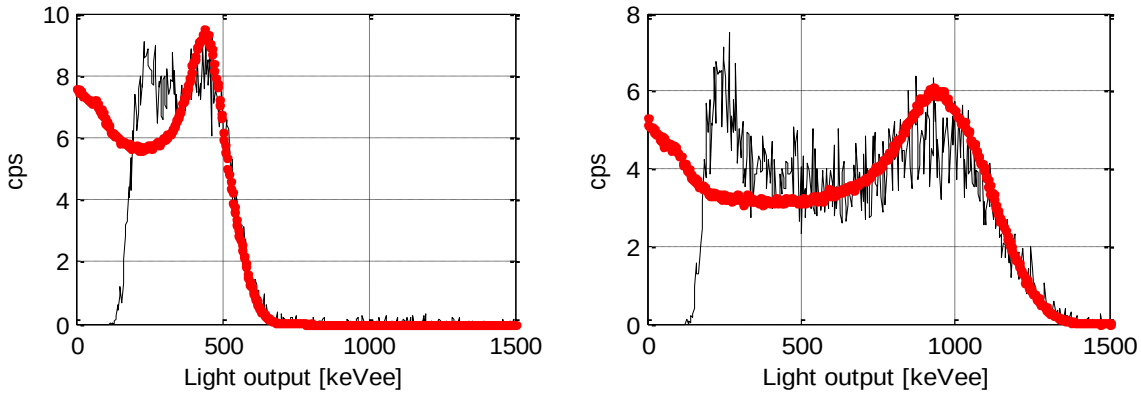


Figure 4. Measured (black line) and simulated (red dots) pulse height spectra for a $^{137}$Cs source (left) and a $^{60}$Co source (right).

A similar overestimation of the detection efficiency of about 15% is observed in Figure 5 which compares the detector response for neutrons resulting from a $^{252}$Cf source.

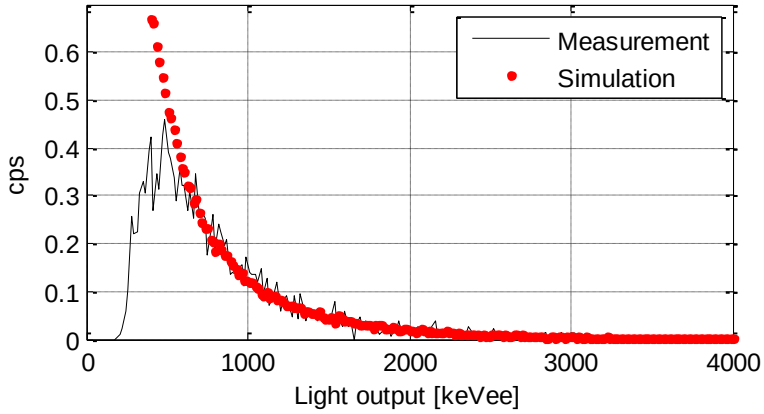


Figure 5. Measured (black line) and simulated (red dots) neutrons pulse height spectra for a $Cf_{252}$ source.

This bias on the simulated detection efficiency is probably due to an overestimation of the effective scintillator volume of the cell. To clarify this overestimation and fully characterize the detector measurements with AmBe and AmLi sources are scheduled.

### 3.4 Energy threshold

A fundamental parameter which characterizes the response of liquid scintillators is the detection energy threshold. This is an adjustable parameter of the electronic settings which discriminates between the accepted and rejected pulses (i.e. the minimum detectable energy). The pulse height spectrum shown in figures 4 and 5 were taken with the minimum electronic threshold setting to minimize noise. The final threshold used in the counting experiments corresponds to the minimum light output where the calculated spectrum still agrees to the experimental values.
As can be noted in Figures 4 and 5, the threshold of the electronics used does not result in a sharp edge on the left side of the pulse height spectrum, but it rather presents some broadening effect. This effect is due to a systematic behavior of the acquisition electronics, which introduces a bias in the threshold determination. Further developments in the acquisition electronics are planned in order to address this issue.
In this work, we derived the energy threshold value with a different approach, that is by matching the total measured counts with successive integrations of the simulated counts from the maximum light output down to a low level that was decreased progressively. Considering the pulse height spectra for $^{252}$Cf reported in figure 5, the value that matches the counts, and thus validates the simulations for the neutron intrinsic efficiency of the detector, was found to be 385 keVee, corresponding to a neutron energy of approximately 1.6 MeV. For the investigated detector, we observed quite poor performances at low energies, meaning that the minimum detectable energy is, by default, very high. Further adjustments in the electronic settings can only be applied to increase this threshold. However, this limit is strongly related to the electronics that is used and does not correspond to the lower limit due to the intrinsic detector characteristics. A possible solution to decrease the lower limit is to either use electronics with different dynamic ranges or increase the high voltage and use multiple outputs at different stages of the photo-multiplier tube.

## 4. Coincidence simulations

Simulations were performed with the MCNPX-PoliMi code [5], a Monte Carlo modeling tool developed to simulate detectors response. Given the physics of liquid scintillator detection, each single collision that occurs in the detection volume needs to be evaluated separately. Depending on the incident particle type and the target atom, the energy deposited is converted into scintillation light with the detector-specific light conversion formula. The amount of light produced by subsequent collisions occurring within a specific pulse-rise time are summed to generate pulses which are then processed by the PSD electronics.

The JRC in Ispra developed a code for post-processing the collision list output of the MCNPX-PoliMi simulations. This code implements the energy-to-light conversion, the PSD discrimination, and the coincident events within multiple detector cells.

### 4.1 Simulation validation on a two-cell coincidence system

In order to validate the coincidence simulation algorithm, measurements were carried out with two liquid scintillator cells, a $^{252}$Cf source placed at 10 cm from each cell, and increasing thicknesses of lead shielding to study the effect of gamma pile-up. The second cell used for these measurement was a cylindrical 5'' x 3'' EJ-309 scintillator. For the simulation of the cylindrical detector response, the light output function was taken from literature data [4] whereas its energy threshold was set to 155 keVee.

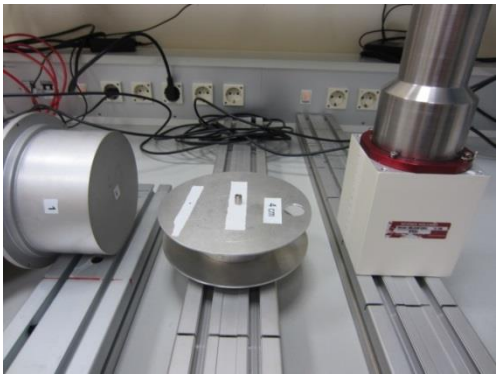

Figure 6. Experimental configuration for the coincidence validation measurements with $Cf_{252}$. The source to detector distance is 10 cm.

Results of the validation measurements are shown in figure 7. While total count rate of the order of 5-10% is overestimated in the results (and is explained by the effective scintillator volume overestimation of par. 3.3), the real count rates are underestimated by a factor of about 20%. This inconsistency in system response is further analyzed in a wider range of neutron energies with coincidence validation experiments using Plutonium oxides and AmLi sources to evaluate cross-talk effects.

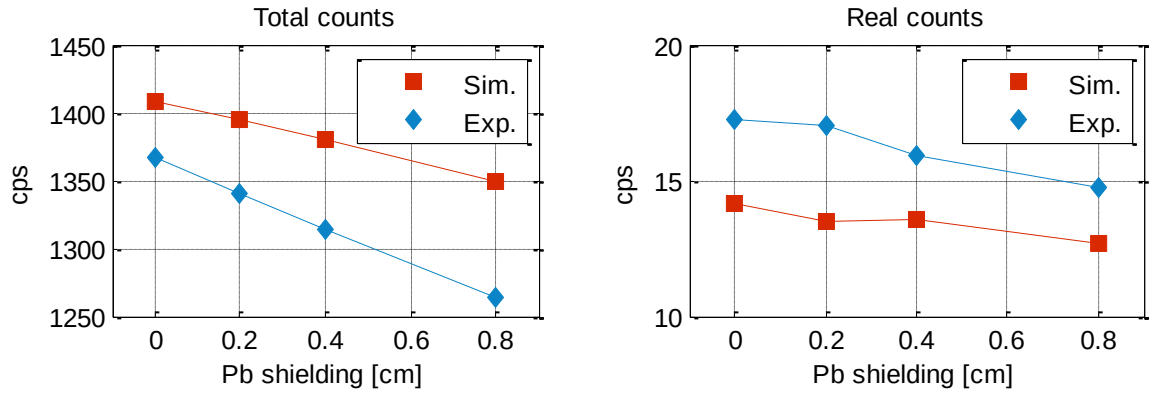


Figure 7. Measured and simulated totals (left) and reals (right) counts rates for the coincidences validation measurements with $^{252}$Cf and different lead shielding of different thicknesses.

### 4.2 Full-system design optimization

Figure 8 (left) shows the proposed design of the prototype liquid scintillator based coincidence collar. It comprises 12 liquid scintillator cells of approximately 1 liter volume, arranged on three sides of the collar. The fourth side is designed to accommodate an AmLi interrogation source. The cells and the source are embedded in a 10 cm thick high density polyethylene (HDPE) wall to increase moderation and multiplication in the fuel element. A 1 cm HDPE interspace between neighboring cells is foreseen to reduce cross-talk events. The internal cavity accommodating the fuel element is surrounded by a 4 mm layer of lead to reduce the gamma rate at the detectors and by a removable 1 mm thick layer of Cadmium used to switch the system to a "fast" configuration when measuring fuel assemblies with

neutron poison. The fast configuration significantly reduces the neutron poison effects by absorbing interrogation neutrons below the cadmium cut-off energy of about 1.25 eV.
Extensive MCNPX-PoliMi simulations were performed to optimize the system response and the design. For this study, a typical PWR fuel element has been modeled. The data of the modeled fuel as well as the intensity of the interrogation source were taken from [6]. Figure 8 (right) shows a section of the simulated geometry perpendicular to the fuel element length.

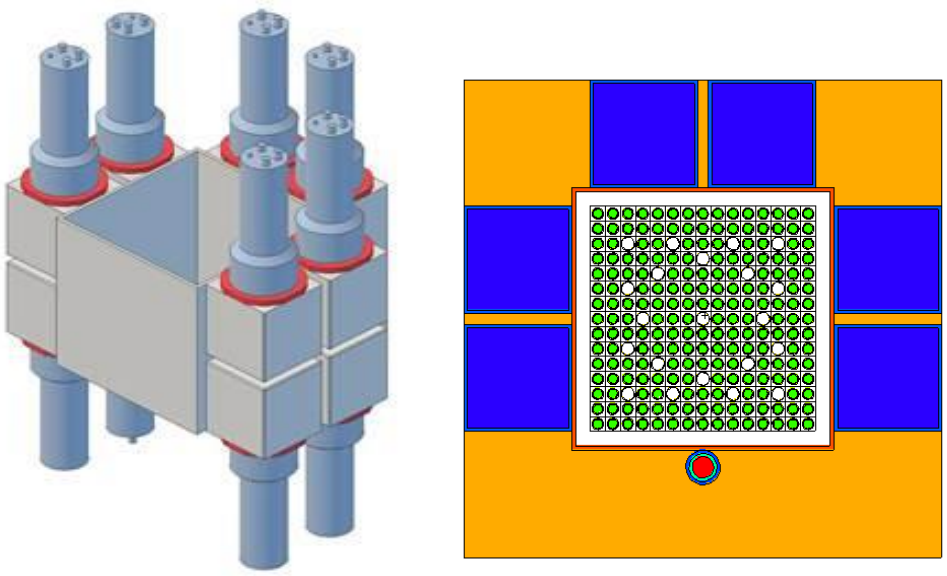

Figure 8. Prototype design of the liquid scintillator based coincidence collar (right) and simulated geometry with MCNPX-PoliMi (left), containing reference fuel element and AmLi interrogation source.

In thermal neutron coincidence counters, the counts rates of the Totals and Reals are dominated by the Accidental count rates which are on their turn driven by the interrogation source. This effect leads to poor statistics and thus long measurement times which are necessary in order to achieve the 1% uncertainty due to counting statistics. One intrinsic advantage of liquid scintillators is that the exceptionally short coincidence time gates given by the detection of fast neutrons leads to smaller accidental rates. To give an example, the coincidence time gates in liquid scintillator is of the order of tens of nanoseconds, whereas tens of microseconds are needed in $^{3}$He systems as to account for the neutron thermalization process.
A first analysis was performed to determine the interrogation source background (no fuel element in the detector) for three different configurations: a bare configuration (no surrounding HDPE); a thermal configuration (10 cm thick HDPE walls on each side of the collar); and a fast configuration (1 mm internal slayer of Cadmium to prevent the thermal and epithermal neutrons from multiple scattering between fuel and moderator). The analyses were performed at different energy thresholds to define the optimal operating condition of the collar. Figure 9 shows the results of the simulation.
The bare configuration presents steadily higher influences of the interrogation neutrons. This is given by the absence of HDPE-filled interspace between neighboring detectors, which act both as a support material for the cells, and as a moderator for the neutrons which do not come directly from the fuel element cavity, reducing their probability of being detected by the scintillators.
As a result of this analysis, the bare configuration has been discarded, and two possible operating settings were identified for the following analysis on the thermal and fast configuration:

- 0.5 MeV neutron energy threshold: this operational setting minimizes the influence of the interrogating source neutrons on the count rate of the Reals. This would allow the detector to be calibrated against the count rate of the Reals, as in current $^{3}$He system, minimizing the AmLi source driven Accidentals.
- 1 MeV to 1.5 MeV neutron energy threshold: this operational setting minimizes the influence of the interrogating source neutrons on the count rate of the Totals. We intend to investigate the possibility of calibrating the detector against the count rate of the Totals, which are no longer driven by the AmLi interrogation source, in order to achieve better statistics in shorter measurement time. When using the Totals data, the count rate of the Reals would be used as to verify of the absence of any nearby uncorrelated neutron source intended to bias the measurements results in an unattended operational mode.

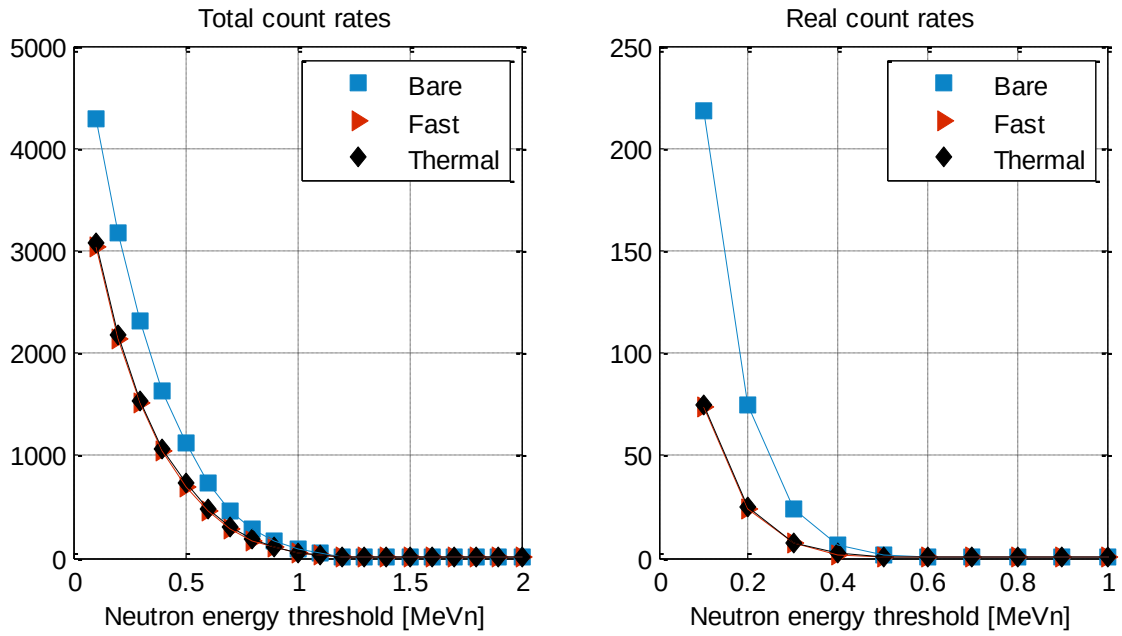


Figure 9. Simulated totals (left) and reals (right) counts rates for the liquid scintillator-based collar in three different configurations: bare, thermal, and fast. The counts are given only by the AmLi interrogation source – no fuel assembly is present. Thresholds are expressed in neutron energy for simplification.

Table 1 shows results of the simulation performed on the neutron coincidence collar for the fuel assembly interrogation, and for two defined configurations and at three different threshold settings. The enrichment of the fuel is 3.19%. The uncertainties resulting from only counting statistics of all the presented data are below 1%.

Concerning the reals count rates, we observed for both configurations that a neutron energy threshold of 0.5 MeV is sufficient to reduce the influence of AmLi neutrons cross-talk to about 1%.

The resulting count rates are then directly related to the induced fissions in the fuel assembly, and a counting statistics uncertainty of 1% can be achieved in a few minutes.

In absolute values, the AmLi source neutrons detected by the scintillators in both configurations are the same, as it results from figure 9. In the fast configuration, however, the Cadmium layer within the cavity reduces the multiplication factor and thus the neutron count rates related to induced fissions in the fuel. Consequently, the relative influence of the interrogation source neutrons is higher in the fast configuration, requiring a higher energy threshold of up to 1.5 MeV (neutron energy) to reduce the AmLi contribution to approximately 1% of the Totals count rate. At this threshold, less than 4 minutes measurement would be necessary to obtain the counts of the Totals at less than 1% relative counting statistics uncertainty. In the same measurement time, the counts of the Reals would be determined with a 5% uncertainty, providing a good indication regarding the absence of any uncorrelated sources nearby the collar which would bias the measurement.

Table 1. Simulation of the system response in the proposed configurations and three different threshold settings for a 3.19% enriched fuel element interrogation. Counting statistics uncertainties of the results are below 1%. Thresholds are expressed in neutron energy for simplification.

| | **Thermal** | | | **Fast** | | |
|---|---|---|---|---|---|---|
| Threshold setting [MeVn] | **0.5** | **1** | **1.5** | **0.5** | **1** | **1.5** |
| Totals [cps] | 1443 | 587 | 313 | 634 | 119 | 51 |
| Reals [cps] | 150 | 39 | 12 | 23 | 6 | 1.7 |
| AmLi totals [%] | 20% | 3% | 0.13% | 70% | 23% | 1.25% |
| AmLi reals [%] | 0.12% | 0.02% | 0.02% | 1.07% | 0.23% | 0.16% |

## 5. Comparison with 3He systems

In the following section we present a comparison of the coincidence collar capabilities with those of a classical $^{3}$He thermal coincidence collar.

We simulated the response of two commercially available $^{3}$He based system, the JCC73 and the JCC71 models, to the same fuel and the same interrogation source intensity simulated in the liquid scintillator analysis. Data were simulated with the MCNP-pta code [7], that postprocess coincidence counters response. The results are presented in table 2.

Table 2. Simulation of the response of two typical $^{3}$He based neutron coincidence collars in thermal and fast configuration for a 3.19% enriched fuel element interrogation.

| | **Thermal** | | **Fast** | |
|---|---|---|---|---|
| | **JCC73** | **JCC71** | **JCC73** | **JCC71** |
| Totals [cps] | 2900 | 2100 | 1200 | 800 |
| Reals [cps] | 190 | 124 | 12 | 6 |
| Accidentals [cps] | 538 | 282 | 92 | 41 |

Considering a classical coincidence methods in which the reals rate is calibrated against the fissile isotopes enrichment, the uncertainty due to counting statistics is given by:

$$\sigma = \frac{\sqrt{2Accidentals + Reals}}{Reals}$$

For liquid scintillators, where the accidental contribution is negligible, the formula can be written as:

$$\sigma_{LS} = \frac{1}{\sqrt{Reals}}$$

Table 3 provides a direct comparison of the achieved counting statistic uncertainty of the two systems for a fast configuration interrogation at different measurements times. Fast configuration is usually more time-consuming than the thermal configuration, therefore it is the most critical for this analysis.

Table 3. Counting statistics uncertainty related to different measurement times for the $^{3}$He based JCC73 and for the liquid scintillator-based coincidence counter, both in fast configuration. Results for the liquid scintillator- based prototype are reported for the Reals mode (threshold set to 0.5 MeV neutron energy) and for the Totals mode (threshold set to 1.5 MeV neutron energy).

| | Measurement time | | | |
|---|---|---|---|---|
| | 60 s | 10 min | 1 h | 10 h |
| **JCC73 σ [%]** | 14.62% | 4.62% | 1.89% | 0.60% |
| **LS Reals σ [%]** | 2.68% | 0.85% | 0.35% | 0.11% |
| **LS Totals σ [%]** | 1.81% | 0.57% | 0.23% | 0.07% |

The advantage of fast neutron detection in liquid scintillators is clearly reflected in the results: it shows a consistent saving in measurement and operational time for the prototype system.

## 6. Conclusions

This paper justifies liquid scintillators as effective $^{3}$He replacement technology for nuclear safeguards applications, given their good neutron detection efficiency and gamma ray rejection capabilities.
We proposed an optimized design for a prototype liquid scintillator-based neutron coincidence collar made of 12 liquid scintillator cells.
The capabilities of the system were analyzed by Monte Carlo simulations using MCNPX-PoliMi code, which proved to be an essential tool for simulating the detector response.
The system modeling required accurate characterization of the liquid scintillators cell response in terms of light output production for different incident particles and energies, and these functions needed to be evaluated separately for each cell geometry.
Finally, a comparison of the proposed prototype with existing $^{3}$He based systems shows promising fast neutron detection characteristics of the liquid scintillator, which is advantageous in terms of measurement time and related statistics, two very important factors in nuclear safeguards applications.

## 7. Acknowledgements

The authors would like to thank Raft Nolte (PTB, Braunschweig) and Peter Schillebeeckx (EC-JRC-IRMM Geel) for their support in the characterization of the detector. The authors would also like to acknowledge professor Padovani (Polytechnic university of Milan, Italy) and professor Pozzi (University of Michigan, MI, USA) for their support during the set-up of the simulations and the post-processor code.

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# Increased transparency in simulations of measurements for nuclear disarmament verification

**Moritz Kütt, Matthias Englert**

Interdisciplinary Research Group Science Technology and Society
Technische Universität Darmstadt
Alexanderstraße 35, 64289 Darmstadt, Germany
Email: kuett@ianus.tu-darmstadt.de, englert@ianus.tu-darmstadt.de

***Abstract:***

*For many safeguards and disarmament verification purposes computer codes are used to simulate and develop different measurements techniques. Often there are some access barriers for the used software (e.g. export controls, high costs for source code). Greater transparency could be provided if a software's source code is made public to everyone dealing with results of calculations. Today, this level of transparency is not required for most applications. But it seems clear that greater transparency of codes will enhance credibility and confidence to foster further steps in nuclear disarmament and non-proliferation.*

*This paper aims to show the utility of the Monte Carlo code framework GEANT for such purposes. Here we concentrate on the simulation of neutron multiplicity counting measurements that will be necessary for nuclear warhead authentication. We simulate the response of a Plutonium Scrap Multiplicity Counter for a Cf-252 source as well as different masses of plutonium. The results will be compared to other simulation codes and measurement results. We analyse differences with the goal to outline possible modifications for GEANT.*

**Keywords:** disarmament verification, transparency, GEANT, multiplicity counting

## 1. Introduction

For any activity in the field of disarmament or non-proliferation increased transparency increases trust and confidence among states and other actors. Such an increase in transparency can be achieved by providing data, opening up facilities to inspectors, having observers present at important processes, and many other means. In the process of verification, declared information is checked by states or other actors, like the IAEA. To ensure that other parties can trust the outcome of verification, this process has to be as transparent as possible too.

During all these described activities computer codes are used to simulate results. In many cases simulations are being conducted during the development and design phase of a measurement device. It is also possible that the execution of measurements could only be done by using simulation codes in parallel. Finally, it might be possible to falsify declarations without doing measurements simply by applying simulations (e.g. calculate possible plutonium stockpiles with reactor burnup calculations). Transparency for all theses types of simulations could be increased provided that a software code would fulfil the following two requirements:

1. Free (no cost, no license limitations) and easy access to the software
2. Free (no cost, no license limitations) and easy access to the source code of the software to ensure “verification of the verification”

Both requirements should be met at least for the actors involved in a verification process. But greater transparency and increased trust could be reached if every interested actor could access software and source code. An additional third requirement could also address simplified ways to guarantee the

correct function of a software.

For software codes frequently used for nuclear verification purposes all these requirements are often unmet. Access barriers can be export controls, restricted source code access or high costs for software and/or source code. Other simulation codes like Geant4 [1] meet at least two of the described requirements – it is available free of charge in source code form. However, a code like Geant4 is less often used for verification purposes so far.

In this paper we examine Geant4 capabilities for neutron multiplicity counting for nuclear disarmament verification. Neutron multiplicity counting is one of the measurement techniques used for the verification of warhead dismantlement and could provide inspectors with the means to verify that a nuclear weapons possessor dismantles its nuclear arsenals.

Increasing transparency increases trust in a verification process. However, in the case of nuclear disarmament verification the level of transparency and the possibility to share information is limited by the necessity for secrecy due to nonproliferation Hence the use of information barriers is often discussed to restrain access to sensitive Information. But their design should be as transparent as possible. Neutron Multiplicity Counting could be used to assess the plutonium mass of a nuclear warhead (an attribute of the item). To help ensuring a transparent development of an information barrier which uses neutron multiplicity counting Geant4 could be useful and versatile tool. This paper examines Geant4 capabilities for the usage for Simulations of Neutron Multiplicity Counting.

## 2. Description of Detector and Simulated Measurements

To test the applicability of the Geant4 for neutron multiplicity counting we modelled the „Plutonium Scrap Multiplicity Counter“ [2], [3]. This detector has been used to estimate the amount of plutonium on scrap materials, but can be used to estimate plutonium masses[1] in general. The detector consists of 80 He3 Tubes in four concentric rings surrounding a cylindrical cavity. The tubes are placed in a High Density Polyethylene (HDPE) block for neutron moderation. The signals are processed by 19 Pre-Amplifiers and later in a multiplicity shift register. The cavity has a diameter of 20 cm and a height of 40 cm. The detector has a very flat axial and radial efficiency profile. We derived the dimensions from drawings in [2] and implemented a model in Geant4. The model can be seen in Figure 1.

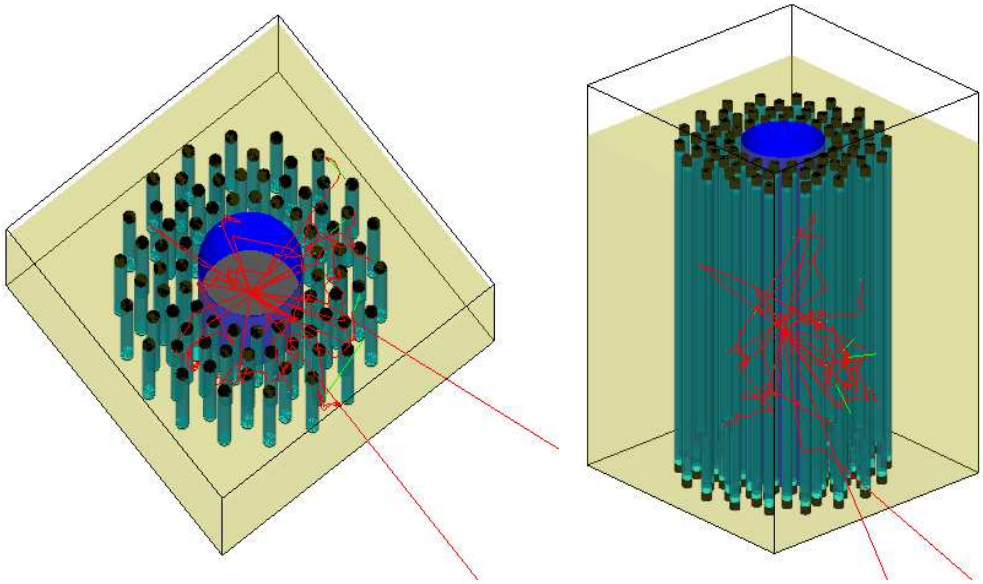

**Figure 1**: Geant4 model of the Plutonium Scrap Multiplicity Counter, view from above and from side. Neutron tracks for 5 spontaneous fission events are shown in red.

1 In fact, it is only possible to estimate the effective Pu240 mass. The total mass can be calculated given that one knows the isotopic composition of the plutonium.

The multiplicity distribution cannot be measured directly due to (necessary) moderation and neutrons originating from other events than spontaneous fissions. Instead, incoming neutron detection events are stored in a multiplicity shift register. After the measurement, the detector electronics directly provides the user with information about the factorial moments of the multiplicity distribution.

Similarly, our simulation has been divided in two parts. The particle transport routines of Geant4 are used to simulate the neutron sources in the detector as well as the neutron transport through the material. Neutrons are tracked until absorption in He3 tubes or other events occur (e.g. leaving the detector geometry). The time information of the absorption processes in the tubes is stored as a neutron pulse train. For the second part. the calculation of the factorial moments, an additional Geant4 library is currently programmed by the authors to carry out pulse train multiplicity analysis in analogy to the analysis carried out by the detector electronics in the PSMC. This library is still work in progress and will be presented elsewhere.

## 3. Code framework GEANT4

Geant4 is a Monte Carlo toolkit, mainly developed for high energy nuclear physics. It is a standard code written in C++ and widely applied in physics research, but rarely used in the field of disarmament, nuclear engineering and reactor physics. To carry out neutron multiplicity calculations with Geant4, mainly three requirements have to be met: good neutron transport capabilities (including thermal neutrons), definition of spontaneous fission sources with correct multiplicities, and capabilities to simulate (α, n) reactions[2].

### 3.1. Neutron transport capabilities

The main requirement to adequately simulate a neutron multiplicity counter is the availability of data for thermal and fast neutron reactions. Geant4 includes data driven models for neutron transport below 20 MeV based on the ENDF-B VII cross-section evaluations, the so-called High Precision (HP) neutron cross sections. These have to be included in the physics list of the Geant4 model. However, by default no data is included in Geant for elements with Z>92. To simulate neutron transport in plutonium we used data libraries (JEFF 3.1, ENDF/B VII) in the Geant4 format, which have been converted by Emilio Mendoza and Daniel Cano-Ott from the Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (CIEMAT), Spain and are available through the website of the IAEA [4]. These libraries also include elements for Z>92. They also fulfill all transparency requirement as they are freely available.

### 3.2. Source Definition for Spontaneous Fission

As important as neutron transport is a correct simulation of the source. To calculate factorial moments from the simulated pulse train the spontaneous fission multiplicity distribution has to be adequately simulated. Geant4 includes capabilities to simulate particles released by the radioactive decay of different isotopes. This is possible also for isotopes with Z>92, if special variables are set. However, by default it does not include spontaneous fission reactions.

Geant4 provides a preliminary hook to include a spontaneous fission source library provided by a group at the Lawrence Livermore National Laboratory [5]. This special library for nuclear fission in Monte Carlo calculations includes data on spontaneous fission multiplicities for various actinides, from which the actual number of particles for each fission can be sampled with a user defined random number generator. It also generates the energy distribution of the released neutrons and gammas in a spontaneous fission event.

To these functionalities we added the capability to use a combination of different isotopes as a single source, which will be necessary for material compositions consisting of different plutonium isotopes and supposedly other actinides. We also added a routine to combine the spontaneous fission source

2 If oxygen or other low-z elements are present these reactions contribute significantly to the neutron background because of the α decay of plutonium.

with the standard Geant4 radioactive decay source, so that alpha decay and spontaneous fission can be used as a combined source.

As the pulse train multiplicity analysis depends also on the specific timing of neutron captures, it is important to adequately sample source events during a specific time interval (e.g. the real measurement time). We sample the time of a source event by using the uniform random number generator of Geant4 and the following function, in which *u* stands for the random number uniformly distributed between 0 and 1 [6]:

$$t = -\frac{1}{\lambda}\ln(\beta + u(\alpha - \beta))$$

with

$$\alpha = \exp(-a\lambda)$$
$$\beta = \exp(-b\lambda)$$

In those formulaes, *a* and *b* describe the time interval (e.g. 0 and 900s for a measurement duration of 900s) and $\lambda$ is the decay constant of the respective decay.

### 3.3. Simulation of (α, n) Reactions

The third capability that Geant4 should meet to yield good results to simulate neutron multiplicity measurements is the integration of (α,n) reactions. Geant4 comes with a model to simulate these reactions with low-z materials, the Binary Light Ion Cascade Model.

In other transport codes like MCNP there is no direct implementation of (α,n) reactions. MCNPX can transport alpha particles but lacks standard (α,n) cross section tables or models. One can only implement an additional neutron source definition based on precalculated neutron spectra from codes like SOURCES4C [7]. In MCNP-PoliMi [8] neutron production from (α,n) reactions is implemented by special sources for oxides of different plutonium isotopes, for $AmO_2$, AmLi, and AmBe mixtures.

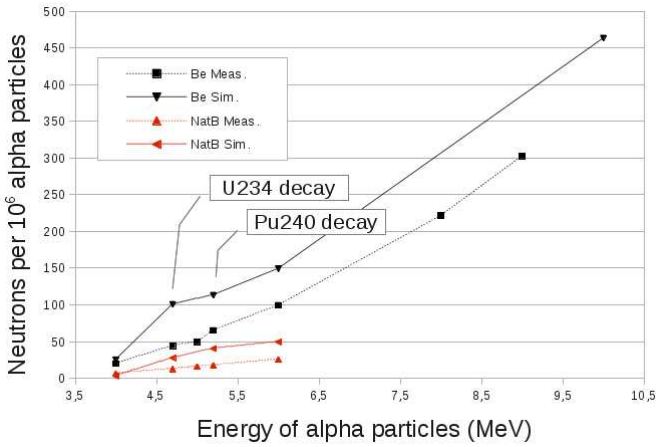


**Figure 2:** Comparison of Geant4 simulated (α,n) with measurement results in [9,10]. The lines are only drawn to guide the reader.

We simulated the reactions with Geant4 using a large sphere of material with an isotropic neutron source in the center. We counted the number of neutrons that were produced by the bombardment of α particles. Figure 2 shows the results for different α energies and for spheres of beryllium and boron in its natural isotopic composition. The measured values were taken from [9,10].

The simulated values reproduce fairly good the energy dependence of the measured values. However, the total numbers are overestimated roughly by a factor of two. For α particles with an energy of 5.2 MeV which corresponds to the energy of particles produced by the decay of Pu240, measurements for a beryllium target show the release of 65 neutrons per million α particles. The simulations show 110

neutrons per million α particles. Similar results were obtained for other energies and materials.
This leaves room for future improvement, by either circumventing the simulation of neutron production by adding a specific neutrons source based on the measured values for every material that undergoes the reactions or to improve the Geant4 model for (α, n). The latter possibility would increase the applicability of the code for many different other problems.

## 4. Detector Efficiency for Cf-252 measurements

With the model of the PSMC and the spontaneous fission source as specified above, we calculated the detector efficiency for a Cf-252 source at different axial positions in the detector cavity. The results are shown in Figure 3.

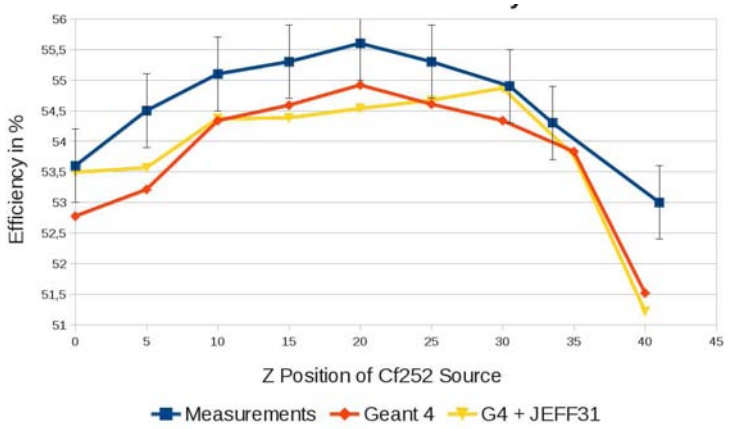


**Figure 3:** Comparison of detector efficiency measurements [11] with simulation based on two sets of neutron cross section based on Geant4 and [4].

We did two sets of calculations, using the neutron reaction data coming with Geant4 as well as the data provided by Mendoza et al. [4]. In general, both data libraries show similar results – the differences can be taken as a measure of the statistical error produced by the simulation. Both calculations show a slight underestimation of the detector efficiency compared to the measurements. [11]. But given that the statistical error of the simulations is around 0.5 %, it is possible to conclude that the simulations are in reasonable accordance with the measurements.

## 5. Conclusion

Transparent software tools are required for non-proliferation and disarmament verification applications. Geant4 can fill in here, as other simulation codes generally do not meet the two requirements for transparent simulation software: free and easy access to the software and the source code. The third requirement – verification of the function of the code itself – will remain difficult and is a field for further development.

For Geant4 we have shown that it is generally feasible to simulate neutron multiplicity measurements using this code. It is possible to simulate different source materials and the particle transport. The necessary neutron reactions are implemented very well, especially using the cross section data provided by Mendoza et al. The deficiencies found for (α, n) can be overcome by implementing special neutron sources for the reactions or by improving the Geant4 reaction model in the future. We started to develop a pulse train analysis class that already shows good results, but will need more development. Further research also has to be conducted to validate the simulation capabilities, by applying the pulse train analysis library and by simulating other detector types and probe geometries.

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# Seismic Monitoring of an Underground Repository in Salt - Results of the Measurements at the Gorleben Exploratory Mine

**Jürgen Altmann**
Experimentelle Physik III, Technische Universität Dortmund,
44221 Dortmund, Germany

***Abstract:***

*We have measured seismic and acoustic signals from various mining activities in the Gorleben exploratory mine in Germany, underground at -840 m and at the surface, tasked by the German Support Programme to the IAEA, in order to provide basic knowledge on the detectability of undeclared activities. During 7 weeks total nearly all sources of sound and vibration available in the mine were covered, with sensors at several positions and sources at several sites, sometimes with background signals from on-going exploration elsewhere.*
*The peak-to-peak values of vibration velocity, referred to 100 m distance, range from tenths of micrometres/second for a hand-held chain saw via few µm/s to tens of µm/s for other tools such as picking, for vehicles, drilling and sledge-hammer blows. A grader with compactor plates produces hundreds, and a blast shot around one hundred thousand µm/s. The last two sources could be detected at the surface, too, at about 1.1 km slant distance; blasts were even seen at 5-6 km distance. The signal strengths vary by a factor 2 to 5 for similar conditions. Fitted by a power law, the decrease with distance is with an exponent mostly between -2 and -1.*
*Spectra of seismic signals from periodic sources (such as percussion drilling or vehicle engines) show harmonic series. Rock removal, e.g. by drilling, produces broad-band excitation up to several kilohertz. Acoustic-seismic coupling is relevant.*
*Monitoring could be done with an underground geophone "fence" around the repository, e.g. 500 m from the salt-dome margin and possibly in the salt 1 km off the repository. With that excavation by drilling and blasting could be detected with certainty by a simple amplitude criterion. Additional research will look at weaker activities.*

**Keywords:** final repository, salt, seismic monitoring, mining, Gorleben

## 1. Introduction

Because in the case of direct disposal spent nuclear fuel contains plutonium such material should remain under IAEA safeguards even after emplacement in an underground final repository. Underground final disposal of nuclear wastes presents a new challenge for safeguards and has led to proposals of using geophysical techniques and methods for monitoring. During operation, the tasks would include monitoring for creation of undeclared cavities and surveillance of those parts of the mine already filled with refuse for undeclared re-opening. After the emplacement phase, when drifts and shafts will have been closed, and the above-ground parts of the final repository will have been cleared for other uses, the IAEA needs the capability of long-term monitoring for covert access to the mine.
Mining and other underground operations produce seismic vibration directly as well as via acoustic noise. Seismic excitation propagates through the ambient medium and can thus be used to detect activities remotely. The main question with seismic monitoring is whether signals from undeclared activities can be separated from signals from different sources and from other background noise. In the operational phase of the repository most noise stems from the normal activity (mining, transport, filling, etc.), and sensors can be deployed at many sites in the mine. After closure, no sensors and cables can remain in the mine; in this phase sensors may be located closer to the surface, and seismic background, produced by traffic, industry, agriculture, weather, and seismicity proper (near and far earthquakes and explosions) will be much lower.

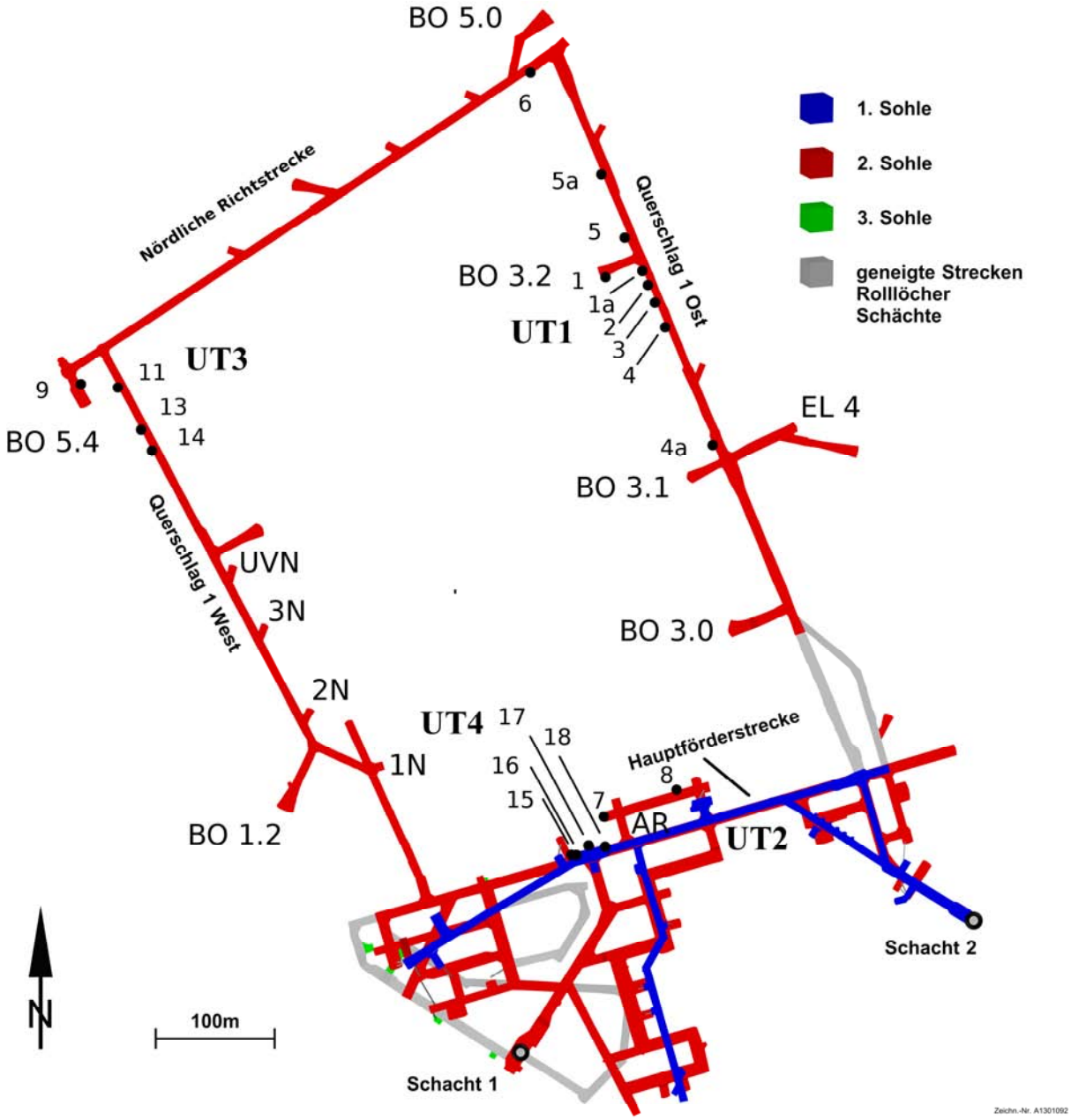


Figure 1 Gorleben exploratory mine, main drift system at 840 m depth, consisting of the "Hauptförderstrecke", "Querschlag 1 Ost", "Nördliche Richtstrecke" and "Querschlag 1 West". Two shafts ("Schacht 1/2") provide access. Indicated are the Dortmund underground measurement stations (UT1, UT2, UT3, UT4) and sensor positions (numbers); not all were used at the same time. (Based on DBE figure)

One potential repository site in Germany is the Gorleben salt dome; it has been explored for its usability since 1986 under contract to the Federal Office for Radiation Protection (BfS) by the firm Deutsche Gesellschaft zum Bau und Betrieb von Endlagern für Abfallstoffe mbH (DBE). Not much is known about seismic signals from mining activities in salt – neither at close range within salt nor after propagation through the surrounding sediment layers. Some seismic measurements had been done on the Gorleben site in the 1990s during sinking of the first shaft, using a temporary geophone network. These measurements were mainly directed at propagation velocities [1]. A network of borehole seismometers around the site (with 5 to 15 km distance) is operated continuously, also by the Federal Institute for Geosciences and Natural Resources (BGR), but this concentrates on the earthquake-safety aspect (e.g. [2]).
In order to gain information on the properties of seismic signals from mining activities, a dedicated measurement project was carried out at Gorleben, tasked by the German Support Programme to the

IAEA. Because seismic signals are also produced by acoustic ones, microphones were deployed, too. Sensors were placed underground and near the surface, at different positions and source distances. Signals from typical mining activities as well as background noise was recorded by geophones, low-frequency accelerometers and microphones. (High-frequency recordings were made in parallel by the Institute for Non-Destructive Testing, Dresden; these are not treated here but will be part of the forthcoming common JOPAG report published by the German Support Programme to the IAEA [3].)

## 2. Measurements

The main measurement systems were deployed in the drift system at 840 m depth (Figure 1). Here nearly all activities took place as well. (Spent fuel would be stored somewhat lower.) In addition a measurement system was deployed at the surface. Sampling rate was 10 kHz throughout, with 16+8 analogue channels recorded underground and 8 at the surface. The underground sensors (some three-dimensionally, others vertically sensitive only) were bolted to the salt-rock wall, at the surface geophones were buried about 0.5 m deep or in 5-11 m deep wells. The stations were synchronised by pulses derived from GPS signals.
After a test experiment of one week in April 2011 the main measurements were done during three weeks each in June/July 2011 and in November/December 2011. Nearly all sources of vibration and sound available in the mine could be measured somehow, but not all of them under good conditions – when which piece of equipment was used where was decided by the operational needs of the mine. In some cases sources were too far to yield useful signals, in others more than one source was active at the same time near the sensors so that one source masked the other. The following main categories were measured: blast shots, vibrating compactor plate, transport installations/activities, drilling rigs, scaler, roof cutter, heavy to light vehicles, hand tools. To produce impulse-type signals a sledge hammer was used for excitation of seismic waves and balloon blasts for acoustic ones.

## 3. Some results

Here the most important results are given. Full detail will be contained in the forthcoming JOPAG report [3].

### 3.1 Wave speeds

Hammer blows were applied to various positions along a drift and recorded by several geophones. From the onsets of the primary (P) and the secondary (S) waves the seismic speeds in the central salt were determined by linear fits to be 4.53 km/s for the P wave and 2.60 km/s for the S wave. The ratio is equal to the theoretical value of $\sqrt{3}$ for a medium where the Lamé constants $\lambda$ and $\mu$ are equal.
Similarly, onset times of the sound wave from balloon blasts resulted in a sound speed of 351 m/s, fitting well to the theoretical value for dry air at 30°C temperature.

### 3.2 Background amplitudes

Seismic background was determined at relatively quiet periods, however with some machinery such as ventilators and the shaft haulage running at far positions. The peak-to-peak values of wall velocity underground varied between positions, from few tenths of a µm/s to several µm/s, for the vertical as well as horizontal components. However, at one position in a drift (Pos. 2 in Figure 1) there was much higher excitation with strong temporal variation (time scale of a second) between few µm/s and nearly 100 µm/s, different from two other positions (Pos. 1a and 5) only 20-30 m away; the reason is unclear. (For background values when the mine was active see also Figure 6 below.)
At the surface the background during quiet periods was between 1.5 and 3 µm/s peak to peak for the geophones at 0.2 m depth and 1-2 µm/s at 7 m depth. During a period of heavy rain the general background increased to about 15 µm/s in 0.2 m and to 3-5 µm/s in 7 m depth, with individual excursions – probably from single big raindrops – between 70 and 150 µm/s in 0.2 m depth.

### 3.3 Variability between sensor positions

Variation between the seismic-signal strengths with position was not only observed with the background, but also with the sources observed. Systematic evaluation was done with signals from picking by an electropneumatic pick hammer (breaker) at one site, recorded after travelling from the western to the eastern drift through about 430 m of salt. Because the picking-caused amplitudes did not dominate the signals, the latter were high-pass filtered with 500 Hz corner frequency; spectral analysis had shown that most of the power from picking was between 500 Hz and more than 3 kHz. Figure 2 shows the peak-to-peak values of the filtered signals during picking after subtraction of the ones befor/after picking at different sensor positions of approximately equal distance, with their respective standard deviations resulting from 5 to 8 usable picking events each. At sensor Positions 1, 1a, 3, 5 and 6 the variation between the events is reasonably low, and the mean values agree between the positions. The higher values at Position 5a are consistent, as are the very low values (around 0.3 µm/s) measured at Position 2. The latter seem to have nothing to do with the very strong low-frequency background at this position.
One can conclude that the seismic-signal strength of the same event measured at various positions of approximately equal distance can vary considerably, here by a factor 0.3 to 1.5. This variability should be taken into account for example when estimating detection ranges. As a consequence signal strengths are mostly presented in logarithmic scale, and decrease with distance needs two or more orders of magnitude to be taken as valid.

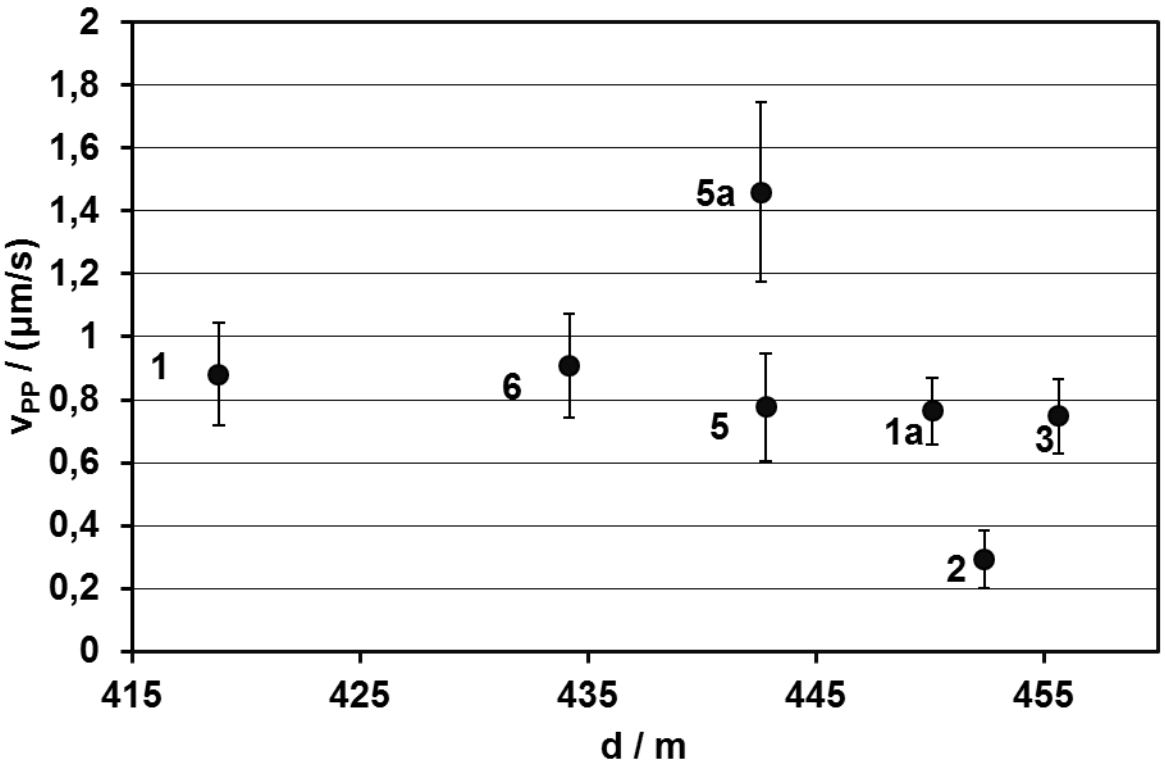


Figure 2 Peak-to-peak value of vertical wall velocity at various sensor positions (indicated by the numbers) of Dortmund station UT1 during eight picking events at the same position (northern corner of "Querschlag 1 West", length coordinate 510 m), versus sensor distance, determined by high-pass filtering with 500 Hz corner frequency and subtraction of the peak-to-peak values before/after picking, in linear scale. Shown are the mean values plus/minus the standard deviations ($N$ = 7 mostly, $N$ = 8 twice, $N$ = 6 and $N$ = 5 once each).

### 3.4 Amplitude decrease with distance

For a seismic volume wave propagating from a point source into a homogeneous medium one expects an amplitude decrease in proportion to the inverse of the distance $r$ due to geometry, times an exponential attenuation term. The present case is much more complicated. The medium is inhomo-

geneous, several wave types may be excited, shafts and drifts can block a direct seismic path but present an acoustic waveguide with weaker decrease of sound amplitude with distance than the 1/*r* dependence valid for free space. Sound impinging on shaft and drift walls can cause them to vibrate; acoustic resonances in the cavities may further complicate the picture. The sources may be extended with a complicated spatial and temporal dependence, they may be driven with different force in varying directions.
Because of the strong signal-strength variation only changes above 30-50% are relevant, and the presentation uses logarithmic scale in most cases. The signal strength is measured mostly by the peak-to-peak value because the root-mean-square value is only sensible for continuous sources. To find a trend a power-law fit to an appropriately selected sub-set of the observed data pairs is used (data scatter is too big to determine exponential attenuation). The empirical power-law exponent of distance thus derived is normally lower than the simple theoretical one of -1.

### 3.4.1 Blast shots

Figure 3 shows three underground signals from a blast at “Bohrort 5.0”. As best seen in the centre trace, there were 12 individual shots with 0.25 s delay. The signal is 4 orders of magnitude above the noise at about 30 m and 2-3 orders at about 200 m. Figure 4 shows the signals at the surface after propagation through about 700 m of salt and 400 m of overlying sediment; here the individual shots are no longer discernible, but the signal is still 1-2 orders of magnitude above the noise. Blast shots were even visible at 5-6 km distance at the seismometers in 300 m boreholes (with accordingly less background noise), here the signal-to-noise ratio (at much lower bandwidth) had decreased to 1-3.
Systematic amplitude evaluations were done with the blast of 1 December 2011 (several shots distributed over about 2 seconds) and a contour blasting of 6 December 2011 (one single shot only). Figure 5 shows the maximum peak-to-peak values of vertical wall or ground velocity. A clear trend can be seen from 140 to 1100 m, here the power-law exponents are -2.3 and -2.1, respectively, much below the theoretical value of -1. The sensor at 45 m distance is at the opposite wall of the adjoining drift, so that it was in the seismic shadow of the drift. The signals from 5 and 6 km have much narrower bandwidth, so they fall outside of the trend, too.

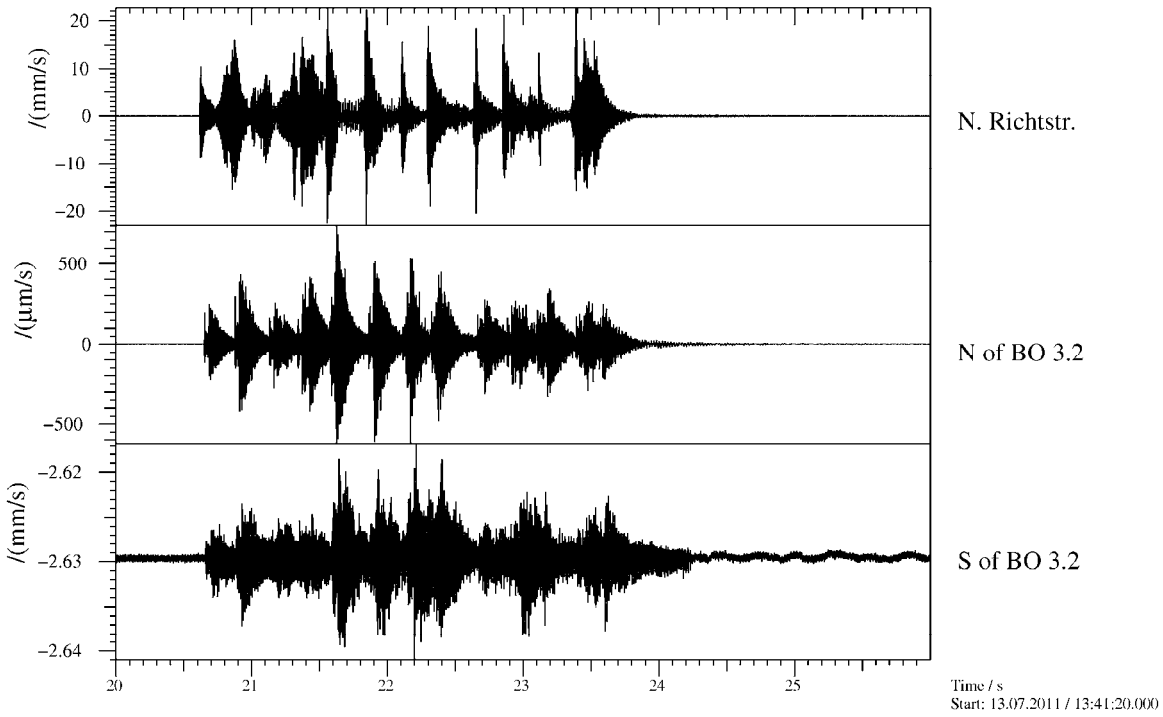


Figure 3 Signals of vertically sensitive underground geophones near site “Bohrort 3.2” during the blast at site “Bohrort 5.0” of 13 July 2011; the blast occurred at 13:41:36.26 CEST. (The PC time given on the time axis is 15.65 s less than the actual time.)
Bottom: Position 2, distance to blast 233 m. (Here the amplitude is uncertain due to the high DC offset.) Centre: Position 5 (distance to blast 189 m). Top: Position 6, distance to blast 31 m). The 12 individual shots are best seen in the centre trace.

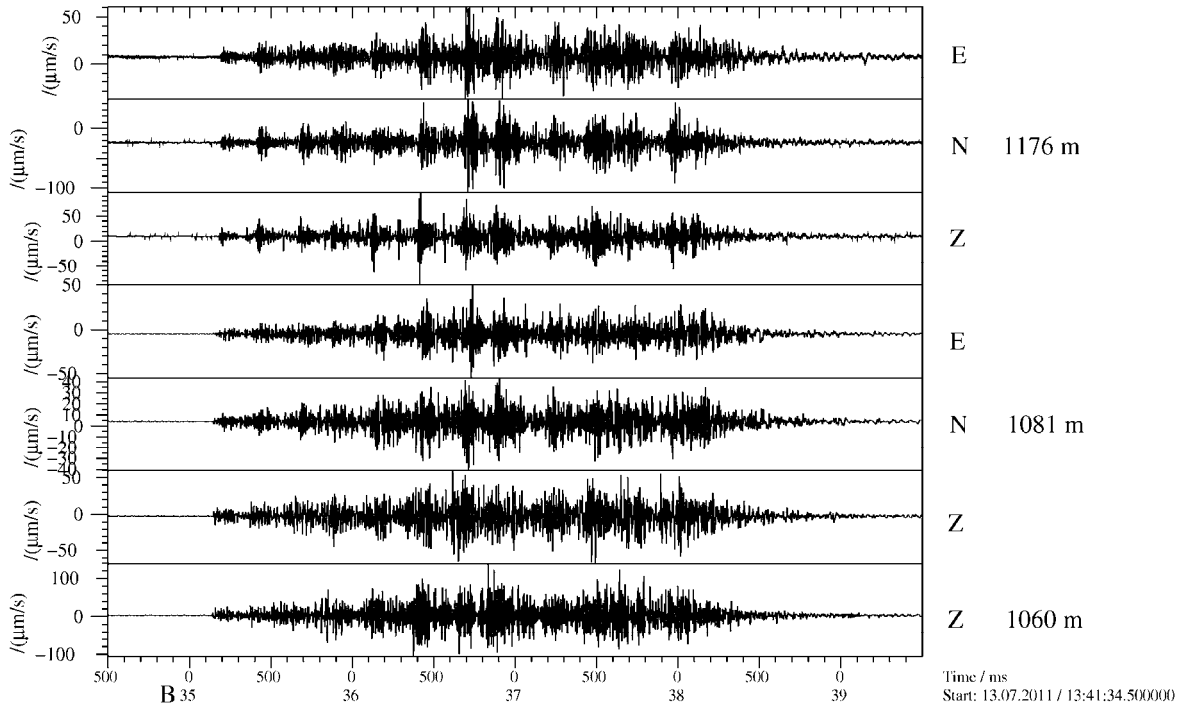


Figure 4 Vertical (Z) and horizontal (N, E) geophone signals at the three positions of the surface station from the blast shot at site “Bohrort 5.0” of 13 July 2011. The blast occurred at 13:41:36.26 CEST (“B”). (The PC time given on the time axis is 1.40 s lower than the real time.) Slant ranges from the blast site to the sensors are given at the right.

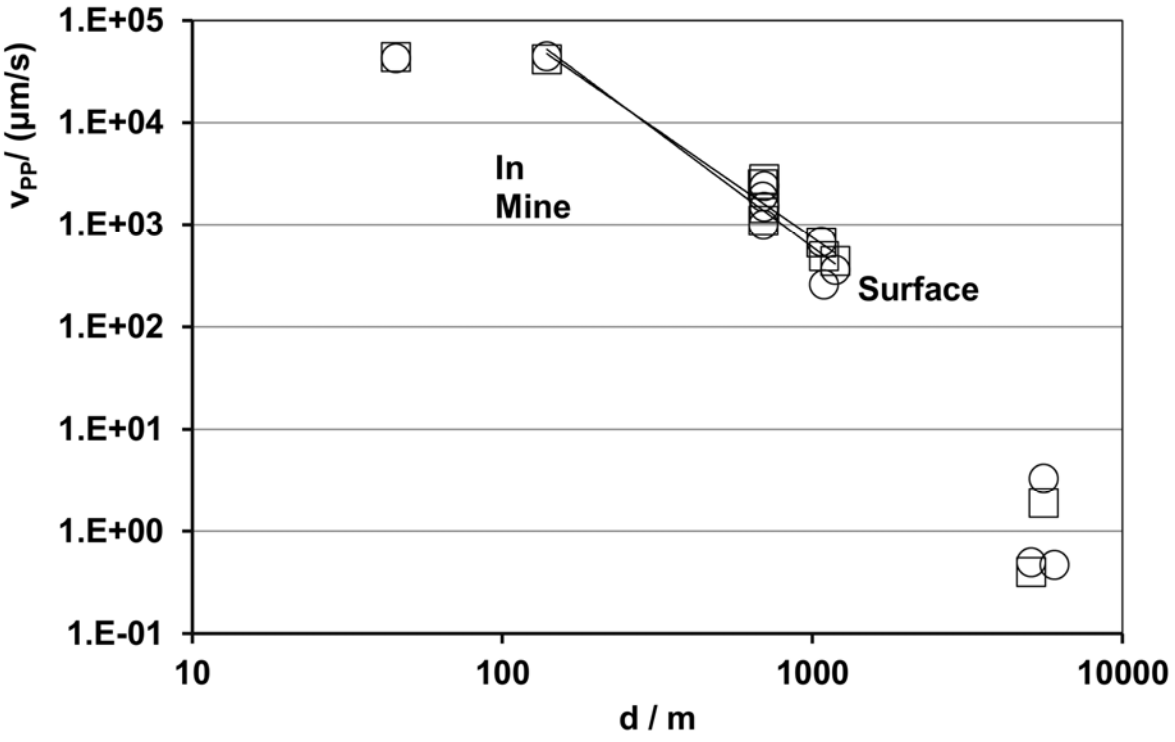


Figure 5 Maximum peak-to-peak value of vertical wall/ground velocity from blasting versus distance for the events of 1 December (several shots, circles) and 6 December (single shot, squares) 2011, double-logarithmic scale. The sensors from 45 m to about 700 m distance were deployed underground in the mine, the ones at about 1100 m were at the surface. The BGR sensors at 5-6 km are in 300-m deep holes, here recording is done with much lower bandwidth. The power-law trend lines from 200 to 1100 m have exponents of -2.3 and -2.1, respectively.

### 3.4.2 Vibrating compactor plate

The grader vehicle at its front carries a plate compactor of three plates (type Stehr SBV 55 H3). Each plate contains two counter-rotating eccentric masses driven hydraulically that cause a vertical vibrating force to the ground (amplitude 60 kN per plate, frequency 60 Hz). This was the second-strongest seismic source, 2.5 orders of magnitude below the blasts. On spectrograms it was also seen at the surface, at 1.1 km slant distance (0.7 km through salt plus 0.4 km through the overlying sediment); after removing machinery noise by high-pass filtering with 90 Hz corner frequency it became even visible in signal amplitude. For this source root-mean-square (rms) values of vertical seismic velocity were evaluated over 20 seconds each while the plate compactors were vibrating, and over about the same time period while they were inactive for comparison. The latter values differed considerably between the positions, probably because different machinery (for example for ventilation) was running at different distances from the sensors. Figure 6 shows the rms values of vertical wall velocity versus slant distance from the respective grader position to the respective sensor. (The corresponding peak-to-peak values are higher by about a factor eight.) Also shown are some of the comparison rms values, the variation between positions is evident.
The general decrease of seismic rms value with distance *r* follows a power law with exponent -1.91. Extrapolating the trend one sees that clear detection of compacting by amplitude only should be possible to around 1 km distance for the lower range of background rms values, but only to 500 or 200 m for the higher-background positions. Obviously for achieving maximum detection distance one should use sensor positions with the lowest-possible background signal.

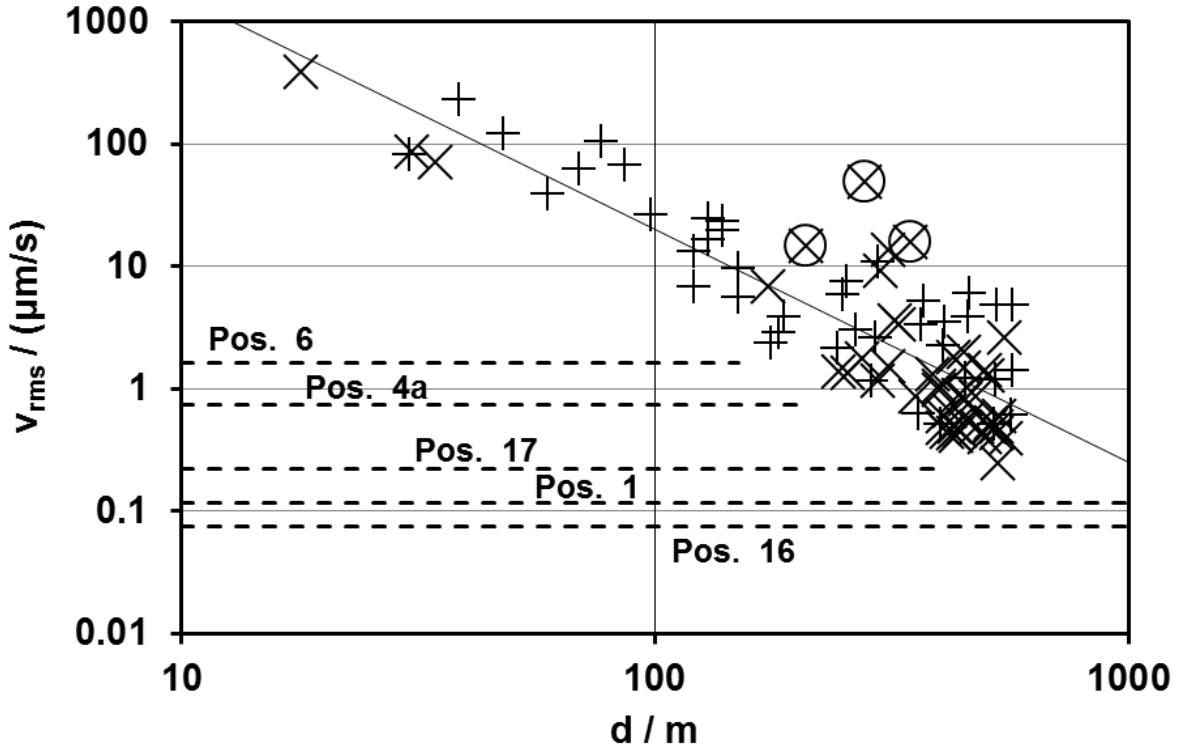


Figure 6 Rms values of vertical seismic velocity during plate-compactor activity versus slant distance from source to sensor in double-logarithmic scale (x: measured at the main Dortmund station UT1 in “Bohrort 3.2”, +: measured at the smaller Dortmund station UT4 in the “Hauptförderstrecke”, near shaft 1). The power-law trend line has a slope of -1.91. (Additional circles mark the signals at Position 6 in the “Nördliche Richtstrecke” from the three sites in the same drift.) The dashed horizontal lines indicate the comparison rms values at several sensor positions when the plate compactors were inactive.

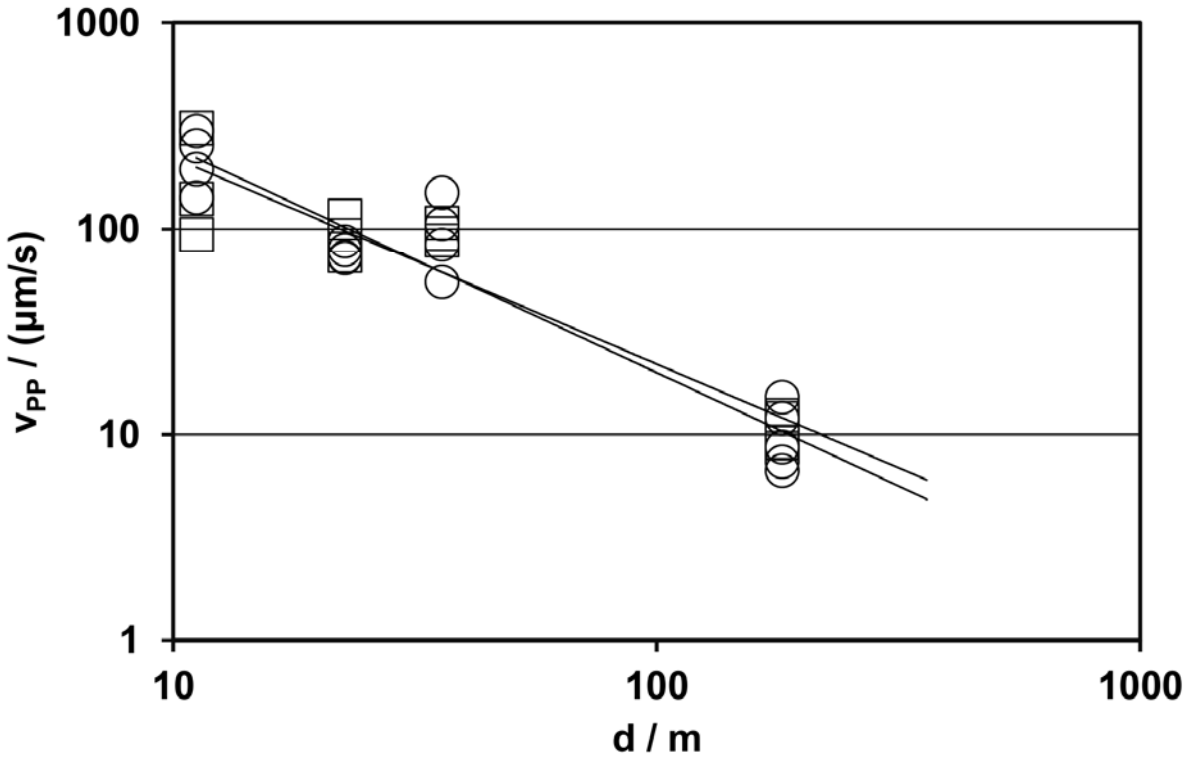


Figure 7 Maximum peak-to-peak values of vertical wall velocity from the scaler versus distance, double-logarithmic scale. Circles: during scaling, squares: in between. For the shortest distance 11 m was used while in fact it varied between about 5 and 15 m by vehicle movement. Power-law trend lines have exponents of -1.1 and -1.0, respectively.

### 3.4.3 Scaler

The scaler is a heavy vehicle with a strong hydraulic arm. At its end there are steel cutting edges at three sides (left, top, right). These edges are moved forcefully along the side walls or roof of a drift to remove salt that has become loose. It is interesting that often the seismic amplitude was higher between the periods of scaling, caused by the engine rotating faster for moving the vehicle or for moving the arm, probably from coupling of acoustic to seismic excitation (see also Figure 11 below).
Figure 7 shows the maximum peak-to-peak values of vertical wall velocity from the scaler versus distance during periods of scaling as well as in between, that is including the high values from engine rotation between scaling. The maxima show considerable variation. Power-law trend lines have exponents of -1.1 and -1.0 for scaling and in between, respectively. Whether these values – differing from the exponent around 2 determined for many other sources – are a consequence of the primary source being acoustic remains to be investigated.

### 3.4.4 Picking

The hand-held electropneumatic pick hammer (breaker) Hilti TE 1000-AVR applies repeated blows to the material via a chisel. For breaking into and removal of salt the chisel angle is varied from time to time by the operator. In order to determine peak-to-peak values at distances above 8 m, a 500-Hz high-pass filter was applied as for Figure 2. Figure 8 shows the results versus distance. The decrease can be approximated by a power law with exponent -1.3, closer to the theoretical one of -1.0 for a seismic body wave without attenuation than the value 2 found with many other sources. However, if only the three points from 8.4 m to 42 m are used (excluding the ones around 450 m), the exponent is -1.79.

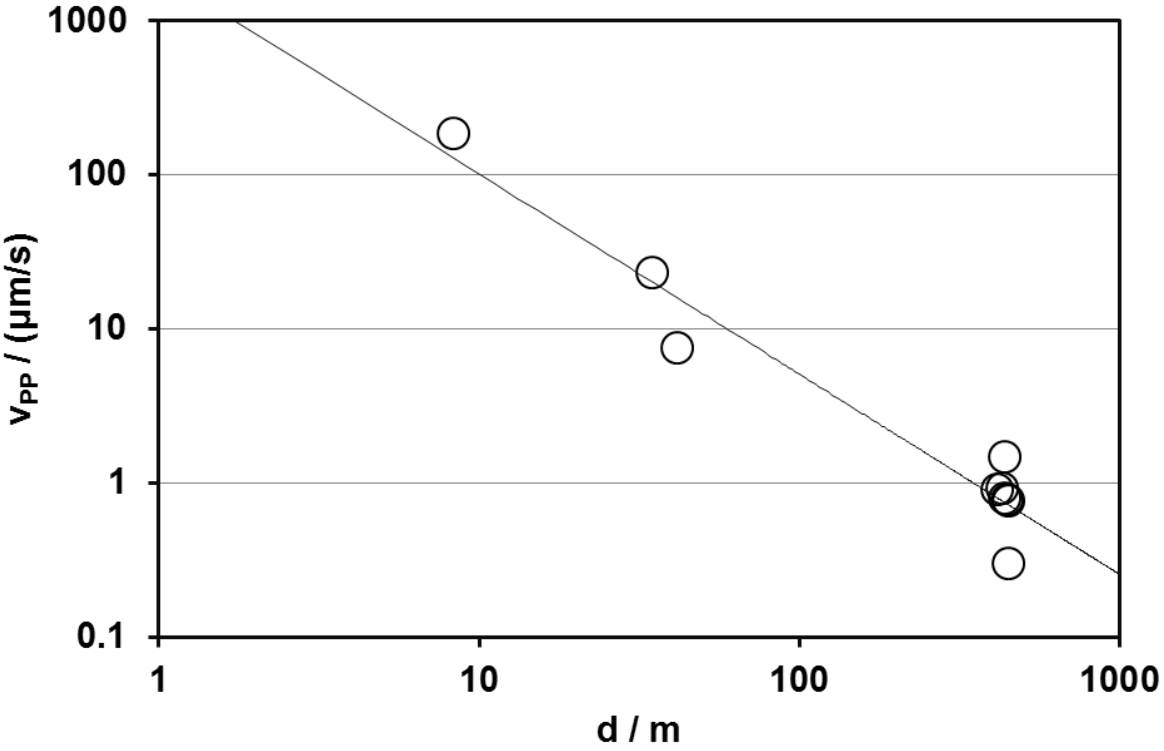


Figure 8 Peak-to-peak value of vertical wall velocity during picking versus sensor distance, double-logarithmic scale. At the lowest distance the signal was evaluated directly, elsewhere it was high-pass-filtered with 500 Hz first. The values around 450 m are the mean values of Figure 2. The power-law trend line has an exponent of -1.30.

#### 3.4.5 Chain saw

The hand-held chain saw (Spitznas no. 5 1029 0010, power 3.0 kW, rotation rate to 6500 $min^{-1}$) is driven pneumatically by compressed air, its blade length is 0.43 m. This turned out to be the weakest seismic source. It was used at two sites in the “Arbeitsraum”; at the farthest, only one sensor showed a discernible signal. Because of masking by different sources (probably a vehicle among them), the recordings were filtered with a high pass of 1 kHz corner frequency, retaining most of the chain-saw-related contributions.
Figure 9 shows the distance dependence of the peak-to-peak as well as the rms values of the chain saw. Trends can only be computed for the closer excitation site, the power-law exponents are -1.99 (peak-to-peak) and -2.07 (rms). These values fit to the -2 found for many other sources, but are less reliable because here the distances extend over half an order of magnitude only. Figure 9 also demonstrates the typical ratio of 8 between peak-to-peak and rms values.

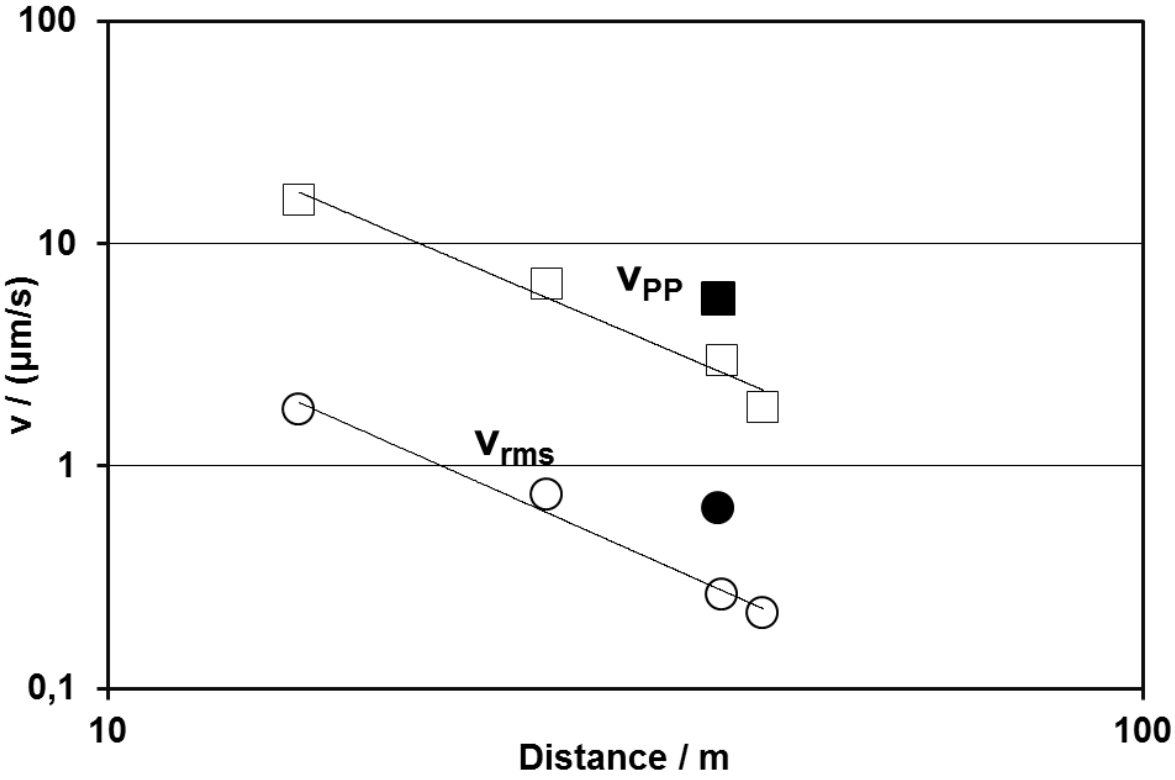


Figure 9 Peak-to-peak and rms values of vertical wall velocity when the chain saw was used, high-pass-filtered with 1 kHz, versus distance, in double-logarithmic scale. Open symbols: sawing at the western wall of the "Arbeitsraum" at closer distances. Solid symbols: sawing at the northern wall with generally larger distances and the drift in the direct path, only visible at the closest sensors; the relatively high excitation raises the suspicion that it was rather caused by a different source. The power-law trend lines for the closer sawing position (open symbols) have exponents of -1.99 (peak-peak) and -2.07 (rms).

### 3.5 Comparison of seismic strengths for various sources

Power-law trends of peak-to-peak values of vertical seismic velocity $v_{PP}$ were determined for many sources. Rough estimates of $v_{PP}$ at some distance $d$ can be gained from this model using

$$v_{PP}(d) = v_{PP}(d_0)\ (d / d_0)^a,$$

using the exponent $a$ found empirically and an arbitrary reference distance $d_0$. In doing so one should keep in mind the variability in signal strength from ±30% to ±50% and more (see Section 3.3). Particular caution is needed when extrapolating beyond the distance interval used for determining the exponent and value at the reference distance.
In this way the power-law trends were computed for 14 representative sources, using the reference distance $d_0$ = 1 m. From this the nominal peak-to-peak value at 100 m distance was computed. This value and the exponent are given in Table 1, together with the range of distances over which the trend was determined. Whenever the range for the trend computation does not include 100 m, the nominal values should be treated with caution, because then they are gained from an extrapolation. Particular caution is also required in case of unusually low or high exponents. Table 1 also contains the general signal properties: pulse, broadband noise or harmonics.

Table 1 Power-law trend data for the representative sources: distance range from which the trend was gained, exponent, ensuing theoretical peak-to-peak value of vertical seismic velocity at 100 m distance, ranked by this value, general properties of signal. When the trend range does not include 100 m, the value should be taken with caution, in particular if the exponent is unusually low or high.

| **Source** | **Range for trend / m** | **Value at 100 m / (µm/s)** | **Exponent** | **Remarks** | **Signal properties** |
|---|---|---|---|---|---|
| Blast | 140-1200 | 1.129E+05 | -2.183 | a | Pulse, repeated |
| Grader compacting | 18-57/90 | 1.906E+02 | -1.906 | | Broadband |
| Heavy loader | 13.2-44 | 7.255E+01 | -0.443 | a, c 1) | Broadband, engine harmonics |
| 5-kg sledge hammer | 7.1-180 | 6.680E+01 | -1.526 | 2) | Pulse |
| Core drilling | 7.1-58 | 3.667E+01 | -0.892 | c 3) | Broadband |
| Scaler | | | | | |
| engine only | 11-180 | 1.998E+01 | -1.098 | 4) | Engine harmonics |
| scaling | 11-180 | 2.198E+01 | -1.005 | | broadband |
| Large-hole drilling rig | 18-99 | 1.791E+01 | -0.798 | | Broadband |
| Hoist platform | 14.3-173 | 1.226E+01 | -1.713 | | Engine harmonics, broadband |
| Roof cutter | | | | c | |
| engine only | | about 3E+00 | - | | Broadband |
| cutting | 16-48 | 1.014E+01 | -1.422 | | Broadband |
| Picking | 8.4-460 | 5.074E+00 | -1.301 | c | Broadband, chisel harmonics |
| Stud driving | 4.6/8.4-42 | 3.008E+00 | -1.756 | c | Pulse |
| Salt through drop hole | 17-24 | 1.887E+00 | -1.798 | a, b, c | Broadband |
| Percussion drilling | 8.4-42 | 1.760E+00 | -1.792 | a, c | Broadband, percussion harmonics |
| Chain saw | 15-43 | 4.100E-01 | -1.985 | c | Broadband |

Remarks: a The closest point(s) are markedly below the trend and were excluded.
b Only two points were used for the trend.
c 100 m is outside of the range for the trend.

Notes:
1) Unusually low exponent.
2) The values from hammer blows at “Bohrort 3.1”, distance 167-210 m, are clearly lower.
3) The closest point (7.1 m) is lower, but is included. Without it (trend from two points) the exponent becomes -1.857, the 100-m value 1.612E+01 µm/s.
4) Too few points for trend were available, but an estimate for 100 m was possible.

The nominal 100-m values are depicted in Figure 10. Assuming a background-noise value one can deduce detection ranges simply from amplitude by defining a detection-threshold signal-to-noise ratio $S/N_{det}$. Somewhat generously requiring $S/N_{det} = 2$, for example with a background peak-to-peak value around 10 µm/s as found during the plate-compactor measurements at Position 6 (8 times the rms value of about 1.2 µm/s, see Figure 6), one concludes that the scaler and stronger sources could be detected at 100 m whereas the large-hole drilling rig and weaker sources would be below the detection threshold. Assuming on the other hand a background peak-to-peak value of 0.6 µm/s as at Position 16 during the grader measurements (the rms value in Figure 6 is about 0.08 µm/s), all sources except the chain saw are detectable at 100 m.

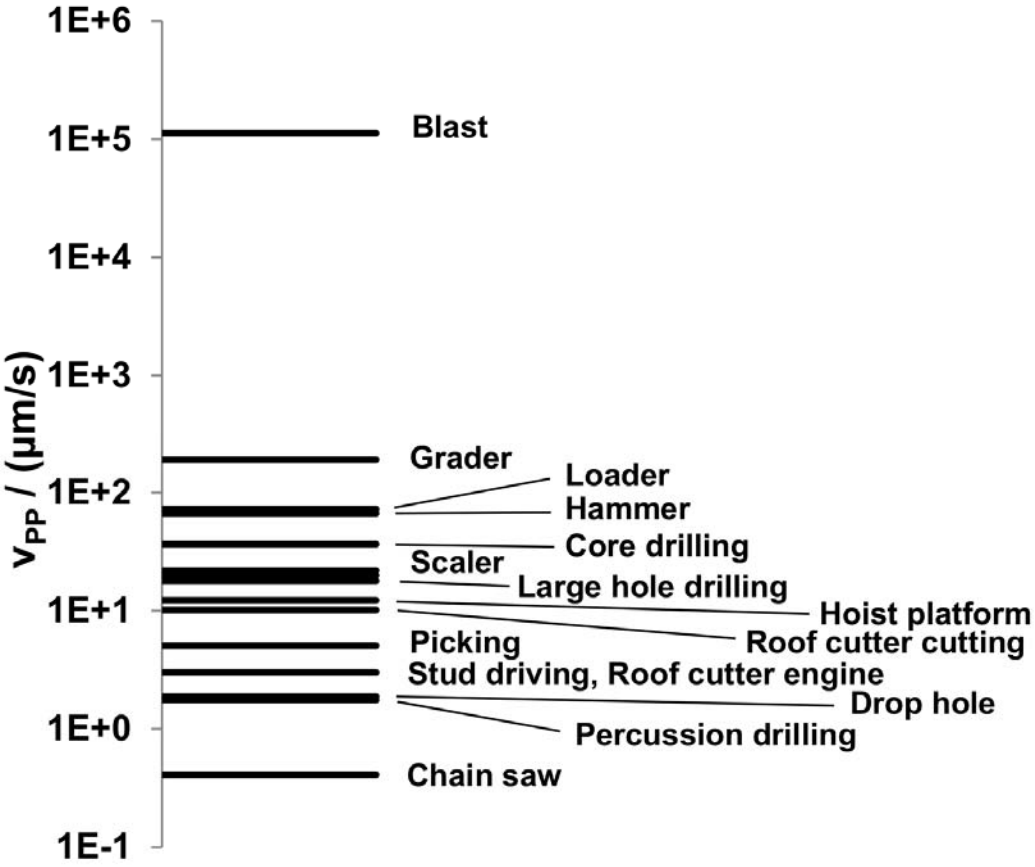


Figure 10 Nominal peak-to-peak value of vertical wall velocity in 100 m distance from various sources, logarithmic scale.

For monitoring after repository closure seismic sensors would be deployed at distances on the order of 1 km. With a quiet mine and sensors at several hundred metres depth a background peak-to-peak value of 0.1 to 0.2 µm/s seems achievable. Summarily assuming a power-law decrease from 100 m to 1000 m with an exponent of -2 as observed with the blasts (see Figure 5) the values at 1 km are lower than at 100 m by a factor 1/100. With a threshold of 2*0.2 µm/s the five sources from blasts down to core drilling would be detectable, with 2*0.1 µm/s the set of detectable sources would expand by the scaler and possibly the large-hole drilling rig. If the signal strength decreases more slowly, for example with a power-law exponent of -1.5, as found for hammer blows, the reduction factor from 100 m to 1000 m is about 1/30. In this case the hoist platform and the roof cutter cutting, maybe also picking would be detectable and the stronger sources would reach out correspondingly farther. More exact statements require looking at the amplitude decrease with distance for each source specifically.
Of course, the amplitude crossing a detection threshold is a simple criterion and not the only one available; looking at spectra or using other more sophisticated detection procedures one can find signals masked by noise.
For seismic monitoring during the emplacement phase – for example to support information provided by surveillance cameras by a second, redundant signal – the background noise will be markedly higher, probably similar to the values measured under different conditions in the present project, resulting in lower detection ranges. On the other hand, sensors could be deployed much more densely, close enough to the potential sources that their amplitude is higher than the background under normal conditions. Here the problem of masking one source by another, stronger one close by has to be tackled.

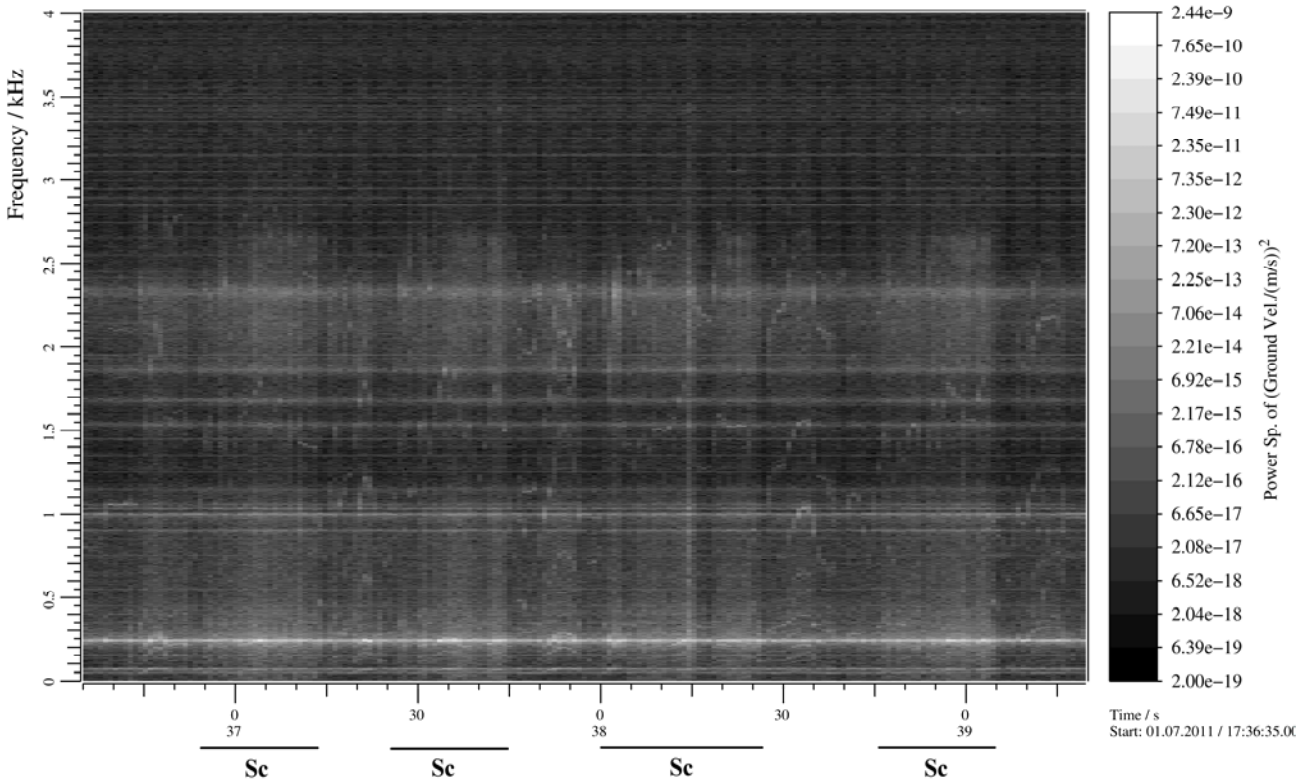


Figure 11 Spectrogram of vertical wall velocity at Position 1 of station UT1 when the scaler worked in the drift close by. Periods of scaling are indicated by “Sc”. In between the frequencies of the engine harmonics increase strongly and decrease again.

### 3.6 Seismic spectra

Here only a few examples are presented as spectrograms.
Figure 11 is a spectrogram of the scaler operating in about 25 m distance. During scaling broadband excitation extends to 2.7 kHz. In the periods in between the frequencies of the engine harmonics increase and decrease again as the hydraulic arm is moved and the vehicle drives to the next scaling position. Acoustic-seismic coupling from the greatly increased engine noise can explain the strong seismic signals in these periods (see Section 3.4.3).
Figure 12 is a seismic spectrogram from a pass of the heavy loader, fully loaded with about 9 t of salt. There is a harmonic series of lines typical of the periodic signals from a reciprocating engine. The frequencies follow the variation of engine rotation rate, jumps can be explained by gear switching. It is notable that there is considerable broadband excitation up to the Nyquist frequency.
Figure 13 is a spectrogram from picking, percussion drilling and stud driving at close range. All spectra contain power nearly up to the Nyquist frequency of 5 kHz. Caused by their periodic blows to the salt rock, picking and drilling show harmonics of a fundamental frequency. This is about constant at 44 Hz for picking, while it decreases with growing friction from about 57 to about 37 Hz as the drill hole gets deeper.

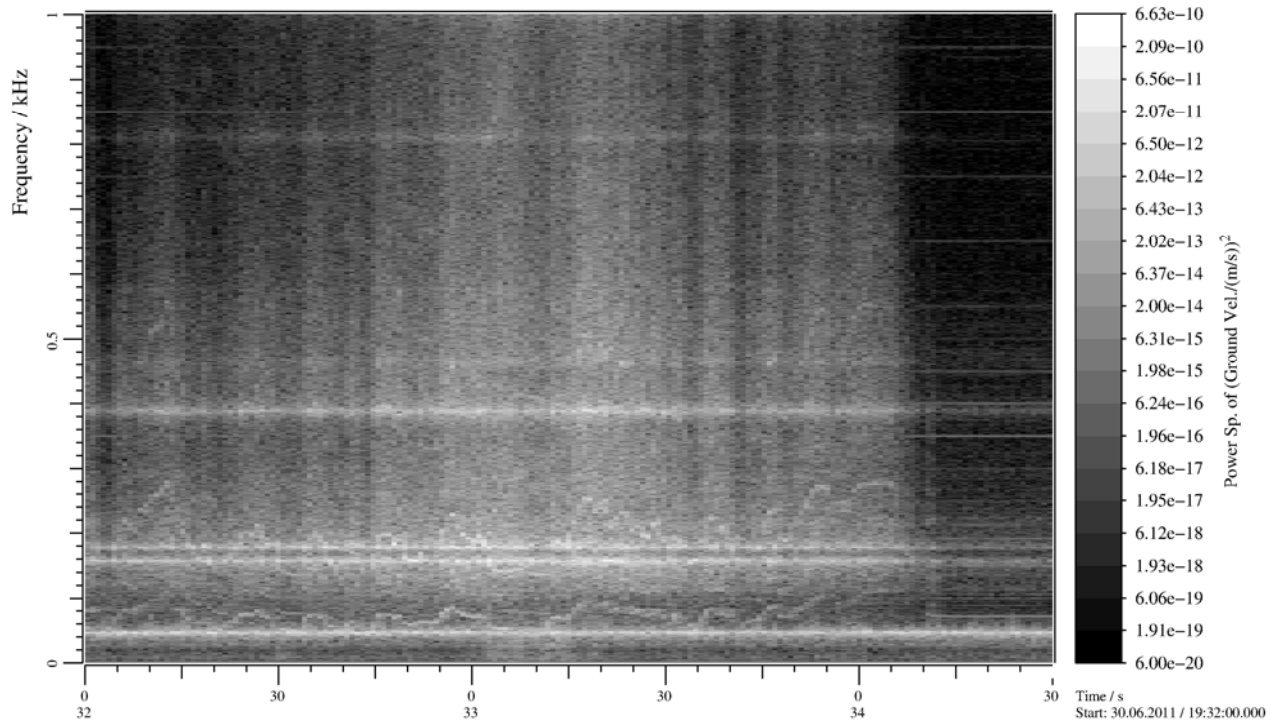


Figure 12 Spectrogram of vertical wall velocity at Position 2 (at the drift wall) during a pass of the heavy loader, fully loaded. The sensor was passed at about 19:33:05.

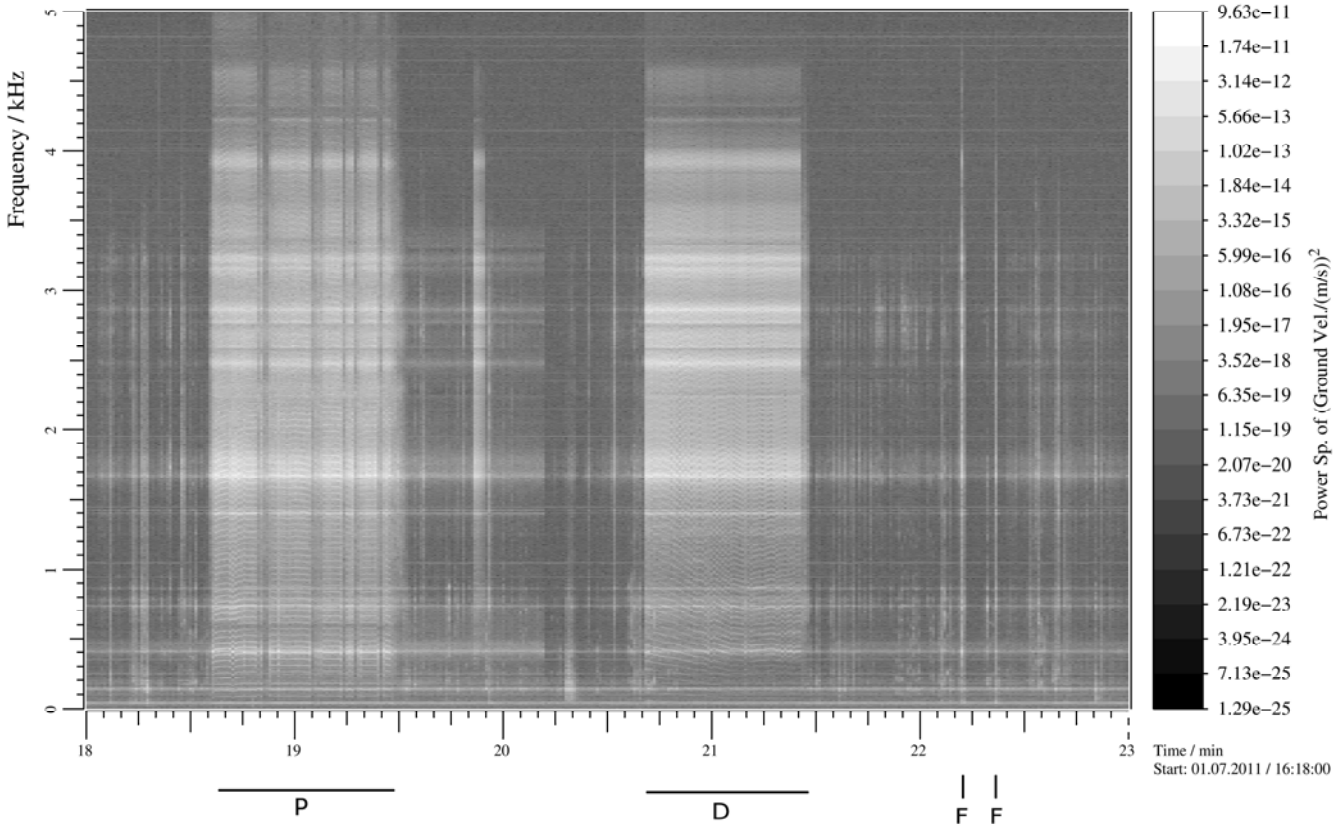


Figure 13 Spectrogram of vertical wall velocity 8.4 m from a site where picking (P), percussion drilling (D) and stud driving (F) was done. Picking and drilling produce harmonic and broadband features, stud driving makes short broadband ones. Several weaker events are visible, too.

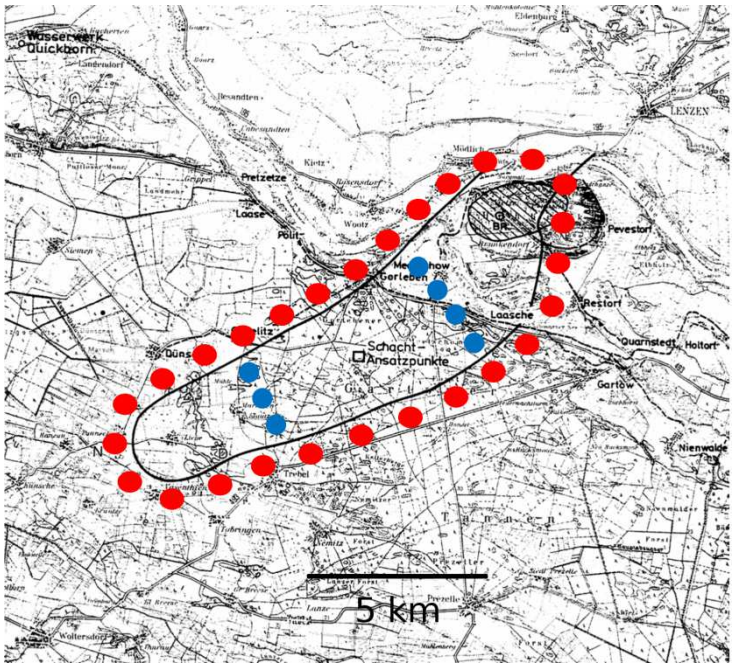


Figure 14 Notional possibilities for placement of seismic sensors after a possible emplacement phase in Gorleben, avoiding the repository volume in the salt-dome centre. Blue: in the salt dome, red: surrounding it. (Based on BfS map)

## 4. Conclusions

On a general level several findings are relevant:

- Removing rock by drilling or cutting produces broad-band vibration.
- Different from drilling the signals from the scaler, but also the roof cutter are irregular, as rock is hit or not during movement of the respective arm.
- Periodically working machinery produces harmonic signals in addition.
- In the underground cavities significant coupling from acoustic to seismic excitation takes place. This is most relevant with the engine noise from heavy vehicles.
- Seismic excitation in the salt rock extends to very high frequencies (tens of kilohertz), and geophone signals contain frequencies up to several kilohertz. In future monitoring at least within the salt dome the high frequencies should not be neglected.

With a view towards seismic detection of mining activities the following can be stated:

- Seismic-signal strength shows considerable variability, peak-to-peak values from the same source at the same distance can vary by a factor 2 to 5.
- Power-law fits to signal strength versus distance give exponents in the range -2 to -1.
- Detection ranges by amplitude alone are many kilometres for blasts, many hundreds of metres for heavy vehicles and, depending on background noise, 100 m to a few times this value, for weaker sources.

The shorter detection ranges are relevant mostly for the emplacement phase when the mine is open and active and seismic sensors could be deployed along shafts and drifts with relatively close spacing, for example 50 m or 100 m. Their signals and the processed results could complement the information gained from video cameras and other traditional safeguards sensors.

After closure of the repository sensors and cables must not remain in it. Monitoring for undeclared activities, that is for excavation in the former mine or close to it, would rely on sensors far from the repository, forming a “fence” in the surrounding sediment, maybe with parts in the salt dome (Figure 14). Keeping 500 m distance from the salt margin and 1 km from the repository in the salt, relevant distances from the repository would be 1-3 km, without sensors in the dome up to 5-8 km. With the mine closed a much lower background will hold. Assuming deployment in boreholes of several 100 m depth, a value of 0.1-0.2 µm/s is plausible. With a power-law exponent of -2, the sources down to the scaler and possibly large-hole drilling could be detected at 1 km distance. This is a very rough estimate – with an exponent of -1.5 instead of -2 the signals at 1 km would be 3-fold higher.

Excavation could be done by two major methods: drilling and blasting, or by cutting machines (tunnel boring machines or smaller road-header machines) (principally, salt could also be removed by water dissolution; if deemed relevant this scenario should be investigated separately). In the first method blasts would provide a very strong and clear signal, even if drilling could not be detected. Removal of rock by cutting machines would set much higher requirements on detection – a road header should be similar to the roof cutter, not a strong vibration source, see Figure 10. But with a road header advance would be relatively slow. A tunnel boring machine that would proceed faster should produce stronger signals with a larger detection range. Such a machine was not available in the Gorleben mine; for this problem at least some estimates should be made.
In terms of monitoring prospects it can be stated that excavation by drilling and blasting can be detected by the sketched concept with certainty. Whether removal of salt by tunnel boring machine or road header machine can be detected at sensor positions far enough away from the repository, and what the requirements on sensors and sensor density would be, needs to be studied.
Most relevant is the investigation of the propagation of seismic waves from sources in the salt dome to longer distances in the salt dome and in particular to the surrounding and overlying sediment. This can be done at first by detailed seismic modelling, including the properties of the various underground media and all boundaries between them. Such a project has begun at TU Dortmund. Some validation of its results will be possible using data gained at the present project, including from the surface. If seismic modelling gives a reasonably positive outcome, then a subsequent project can validate the results at other positions using sensors in a few exemplary boreholes.

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# Fast neutron coincidences from induced fission as a method for detection of SNM

**A. Ocherashvili[a], M. Mosconi[b], J-M. Crochemore[b], A. Beck[a], E. Roesgen[b], V. Mayorov[b], B. Pedersen[b]**

[a]Physics Department
Nuclear Research Center Negev
P.O. Box 9001, 84190 Beer-Sheva, Israel

[b]Institute for Transuranium Elements (ITU)
Joint Research Centre, European Commission
Via E. Fermi 2749, Ispra 21020 (VA), Italy
E-mail: bent.pedersen@jrc.ec.europa.eu

***Abstract:***

*A method for the detection of special nuclear materials (SNM) in shielded containers which is both sensitive and easily applicable under field conditions is presented. The method applies neutron induced fission in SNM by means of an external pulsed neutron source with subsequent detection of the fast prompt fission neutrons. Liquid scintillation detectors surrounding the container under investigation are able to discriminate gamma rays from fast neutrons by the so-called pulse shape discrimination technique (PSD). One advantage of these detectors, besides the ability to do PSD analysis, is that the analogue signal from a detection event is of very short duration (typically few tens of nanoseconds). This allows the use of very short coincidence gates for multiple detectors for the detection of prompt neutrons from the same fission event while benefiting from a low accidental (background) coincidence rate. These features result in a relatively low detection limit of the fissile mass.*
*Another principal advantage of this method derives from the fact that the external neutron source is pulsed. By proper time gating the interrogation can be done by either epi-thermal or thermal neutrons. These source neutrons do not appear in the neutron signal following the PSD analysis thus providing a fundamental method for separating the interrogating source neutrons from the response in form of fast fission neutrons.*
*The paper describes laboratory tests with multiple detectors at the Pulsed Neutron Interrogation Test Assembly (PUNITA) for the purpose of investigating the measurement principle. Results of thermal and epi-thermal neutron interrogations are shown and discussed.*

**Keywords:** security; fission; scintillation; PSD; coincidence;

## 1. Introduction

A variety of passive and active non-destructive assay (NDA) methods are being investigated for the purpose of detecting special nuclear materials (SNM) in practical applications [1-3]. The prompt emissions in fission events of neutrons and $\gamma$-rays are useful signatures for the detection of SNM in shielded containers. One reason for this is the fact that the component of high energy prompt radiation from fission is very penetrating and thus difficult to deliberately shield from detection. Furthermore identifying the detected radiation to be originating from fission events is

evidence of the presence of SNM in the object under investigation. To this end it is useful to arrange the detection system to take advantage of the fact that during the fission event multiple prompt $\gamma$-rays and neutrons are emitted simultaneously [4-5].
Using an external neutron source to induce fission extends the usefulness of this detection method to apply not only to spontaneous fissile materials but also to materials with a cross-section for neutron induced fission. Pulsing of the external neutron source can provide further advantages to be exploited in the detection method. This includes the fact that by proper timing (gating) of the detection period with respect to the neutron emission from the external source the object can be interrogated by a low energy neutron flux (epi-thermal or thermal neutron flux) only, providing the possibility to distinguish the fast fission neutrons from the low energy source neutrons in an appropriately selected neutron detection system [6].
In the present work we investigate the use liquid scintillation detectors for the detection of the prompt radiation from fission events. These detectors can distinguish fast neutron interactions from other interactions by means of the so-called pulse shape discrimination (PSD).
The intention with this work is to demonstrate the principle of the detection method by means of a detection system which can be scaled up, e.g. by increasing the number of detectors, to handle objects a few cubic metres such as the standard universal load devices (ULDs) used for air traffic cargo.

## 2. Experimental setup

The Pulsed Neutron Interrogation Test Assembly (PUNITA) of the Joint Research Centre is designed for experimental studies in non-destructive analysis (NDA) methods for nuclear safeguards and security [7-8]. Figure 1 shows a cross section of PUNITA and a view inside the sample cavity. The facility is composed of a large graphite liner surrounding a central cavity of volume 50x50x80 $cm^3$. The (D–T) pulsed neutron generator, the sample under investigation and the scintillation detectors used for coincident detection are located inside the cavity. In total 96 one metre long $^3$He neutron detectors are embedded in polyethylene modules and shielded by cadmium (fission neutron counters in Figure 1).

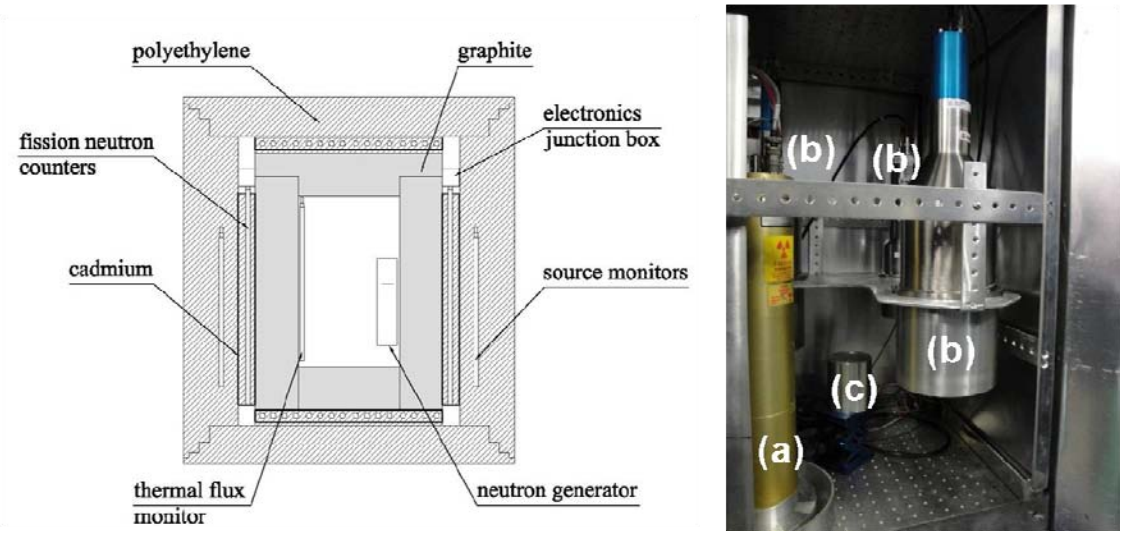



**Figure 1:** Cross-section of PUNITA showing the permanently mounted neutron detectors and the neutron generator (left picture). The picture shows the sample cavity of PUNITA including neutron generator (a), liquid scintillation detectors (b), and uranium test sample (c).

In the present experiments these detectors are used as reference detectors of the prompt fission neutrons. The neutron generator (Model A-211 from Thermo Fisher Scientific Inc.) is pulsed at 100 Hz which is tailored to the average thermal neutron lifetime in the graphite/cavity configuration. The thermal flux generated by source neutrons being thermalized in the graphite peaks at about 250 μs after the 14-MeV neutron burst. The generator is able to produce short and intense bursts of neutrons with absolutely no neutron emission between bursts. This fact, together with the very short duty-cycle of one per mille, allow separation of the neutron interrogation into a fast/epi-thermal period from zero to 100 μs, and a thermal period from 250 μs to 9 ms, respectively. In Ref. 6 the neutron field of PUNITA as function of time is explained in detail.

The scintillation liquid used in this work is the BC-501A from Saint Gobain Inc. Three detectors are used in the present experiment; two detectors of dimensions 5"x5" and one detector of 3"x3". The anode output of the photomultiplier is connected directly to a signal digitizer. The digitizer in this work is a four channel LeCroy WaveRunner HRO66 ZI oscilloscope with a 2 GS/s sampling rate and 12-bit ADC.

The performance of the BC-501A scintillation detectors with respect to n/γ discrimination in the PUNITA facility is described in Ref. 6. Due to the very fast response of such detectors the effect of the neutron generator burst can be followed in detail. This is for example not possible in the $^{3}$He proportional counter based "fission neutron counters" surrounding the graphite liner. The charge collection of these detectors is two order of magnitude slower, and they tend to saturate for as much as 250 microseconds after the neutron generator burst. Figure 2 shows the total detection rate in the 3"x3" detector following a neutron generator burst initiated at time zero. In the time period of 20 to 100 μs after the burst the response decays with a single exponential representing the slowing-down of the fast source neutrons. From 250 μs onwards the response (not shown) is dominated by the decay of the thermal neutron flux.

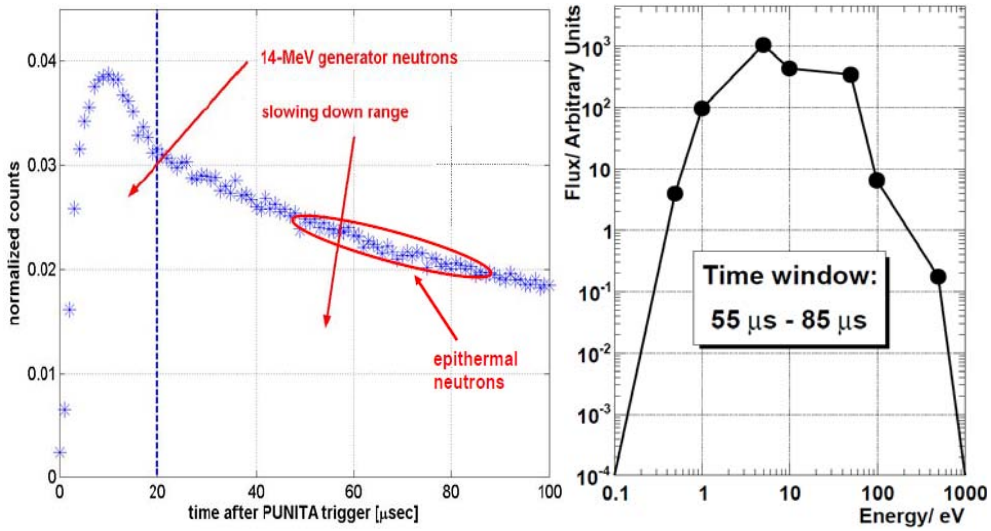



**Figure 2:** Left side: Time spectrum of detector response in PUNITA after neutron burst (at time zero), detector: BC-501A scintillation detector. The time interval when epithermal neutrons persist is highlighted. Right side: MCNP simulation of the neutron flux in the centre of PUNITA in the time interval 55 μs – 85 μs after the 14-MeV neutron burst.

In Figure 2 the region of 55 – 85 μs is highlighted and represents a region where only source neutrons below 500 keV persist. The energy spectrum of the source neutrons as calculated by MCNP [9] is also shown in Figure 2. The neutron energy is mainly in the range 0.5 eV to 200 eV. This interrogating neutron spectrum is particularly interesting as the energy is too low to produce a "neutron response" in the PSD spectrum thus allowing to distinguish the low-energy source neutrons from the high-energy fission neutrons in the detector response.

In the present work we apply a simplified coincidence scheme which will trigger the storage of events when at least two out of the three detectors have observed a gamma or neutron interaction. At each trigger event all three detector channels are stored for off-line analysis. The trigger scheme of the coincidence detection is shown in Figure. 3. Each scintillation detector is coupled to a constant fraction discriminator (CFD, ORTEC 935 Quad 200-MHz Constant-Fraction Discriminator). The walk of each CFD was tested using a $^{60}$Co source and their threshold was set to 0.2 V by means of a calibrated amplitude pulse generator. The width of the output pulse was set to 50 ns to impose the requirement that at least two signals must occur within 50 ns to trigger the storage of the event.
The synchronizing digital output signal of the neutron generator is sent through a variable delay to the veto of the CFDs. This feature allows to select the time period of interrogation with respect to the burst of 14-MeV neutrons from the generator. In this work we investigate two time intervals: the one of Figure 2 (epi-thermal interrogation) and one after 300 μs (thermal neutron interrogation). An ORTEC CO4020 Quad 4-Input Logic Unit is used to implement the two-out-of-three coincidence trigger for the three detectors. The coincidence triggers are summed in a fast OR gate, and used to trigger the storage of the LeCroy waveforms.

Sending a trigger signal to the oscilloscope signifies that a double coincidence event (or triple coincidence event) has occurred. Such a trigger will cause storing of all three detectors channels in the LeCroy internal memory. A waveform of 200 ns duration is stored on each input channel.

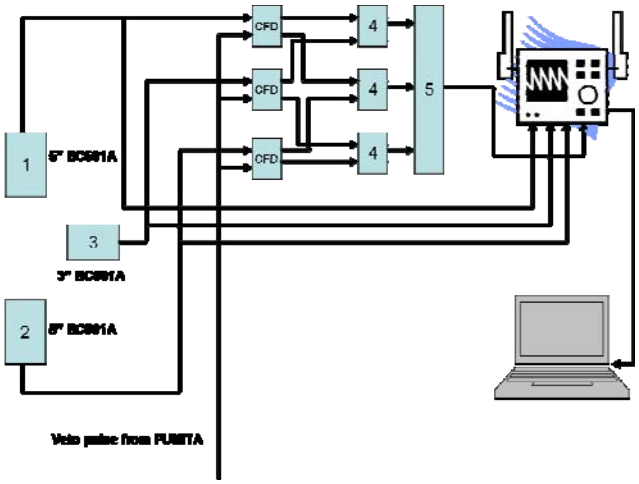



**Figure 3:** Trigger scheme for the data acquisition. The boxes labeled with 4 are coincidence units and the box labeled as 5 is a fast OR unit.

Tens of thousands double or triple coincidence events are stored in each experiment for offline analysis. We use MATLAB to perform the PSD analysis (neutron/gamma discrimination) and plot the coincidences according to their multiplicity (double or triple) and kind (gamma-gamma, gamma-neutron, neutron-neutron, etc.).

The BC-501A liquid from Saint Gobain Inc. [10] produces scintillations according to different mechanisms for neutron and gamma ray interactions. The PSD method utilizes the fact that scintillations caused by neutron interactions result in a longer decay component compared to that of gamma interactions. Various techniques can be implemented to yield the discrimination between fast neutrons and gamma ray signals. Most common perhaps is the so-called integration method [11]. We apply a variation of this method as illustrated in Figure 4. Analytical expressions obtained by fitting of the scintillation detector anode outputs [12, 13] are used to distinguish the waveforms. The difference between the two signals in Figure 4 (a) appears to be largest at 22.8 ns after the start of the pulse. In practice we use the cumulative sum of the normalized waveform as shown in Figure 4 (b). This figure shows a more pronounced difference between the two pulses. At the level of 90% the difference is 40.8 ns. By implementing a simple

algorithm that records the time for reaching the 90% accumulative sum, a speedy and effective distinction between gamma ray and fast neutron detection events is achieved.

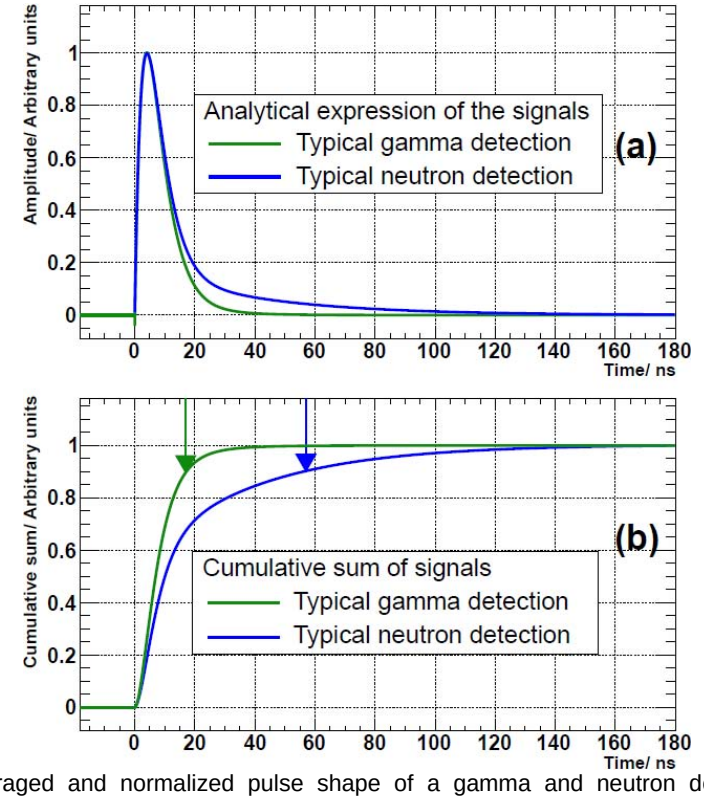



**Figure 4: (a)** averaged and normalized pulse shape of a gamma and neutron detection event. **(b)** Cumulative sum of pulses in (a), also indicating the time of 90% sum used as PSD signature.

## 3. Measurements of uranium samples in PUNITA

A series of standard CNNM $U_3O_8$ sources [14] are used in conjunction with the pulsed neutron interrogation and three BC-501A scintillation detectors. The five CBNM standards are identical in all aspects (total mass, density, geometry, container type) except for the $^{235}U$ enrichment. The mass of the fissile $^{235}U$ component is 0.52 g (0.31%), 1.12 g (0.71%), 3.28 g (1.94%), 4.99 g (2.96%) and 7.54g (4.46%), respectively. Also measurements of an empty CBNM container are included for the purpose of comparison. The sample is placed in the centre between the three detectors at a distance of 150 mm as seen in the picture of Figure 1.

In the following we show results of interrogation with thermal neutrons (300 μs to 4000 μs after the 14-MeV burst) and epi-thermal neutrons (55 μs to 85 μs after the 14-MeV burst). Each experiment includes 30,000 generator pulses (5 minutes measurements). Figure 7 shows the normalized rate of detected coincidences of the types: (γ-γ), (γ-n) and (n-n) as obtained in the offline PSD analysis. The normalization corrects for variations in the neutron emission rate of the neutron generator.

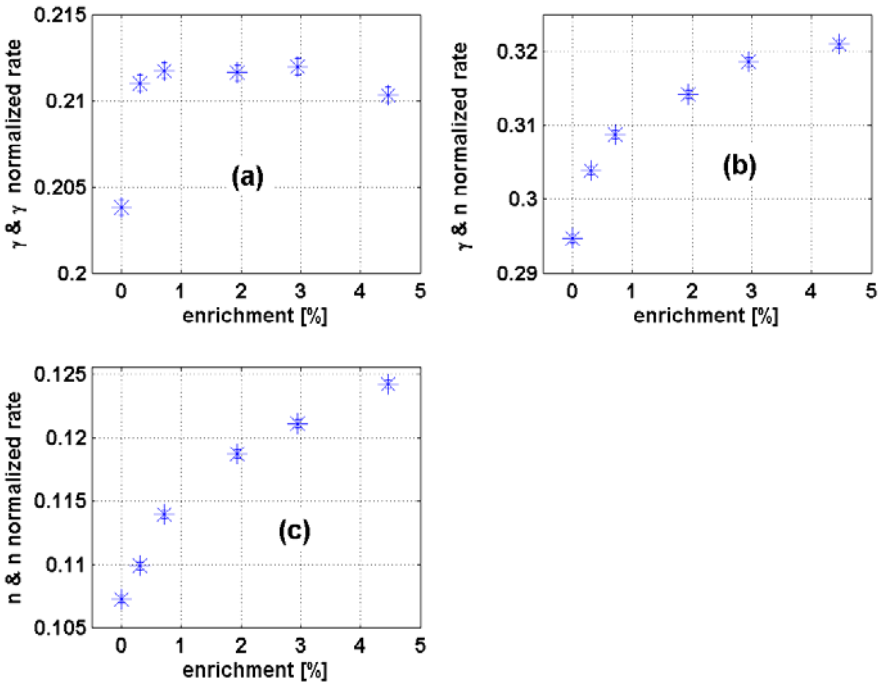



**Figure 5:** Thermal neutron interrogation: coincidences for different particle composition: γ-γ (a), γ-n (b) and n-n (c) for the five CBNM uranium standards and the empty container**.**

The (γ-γ) coincidence rate in Figure 5 (a) shows no dependence on the $^{235}$U concentration. This is believed to be due to the gamma emissions from fission events being swamped by other gamma emitting reactions (e.g. neutron capture). Interestingly the empty container has significantly less emission suggesting that the difference is due to gamma coincidences originating from uranium.

The (γ-n) and (n-n) coincidences, Figure 5 (b) and (c), however show a clear dependence on the $^{235}$U concentration indicating that these coincidences (with a neutron component) originate from induced fission events. The fact that this dependence is not linear is due to the self-attenuation of the thermal neutron flux inside the stronger absorbing (higher $^{235}$U concentration) samples.

A similar impression is seen from the analysis of triple coincidences. Figure 6 shows the different types of triple detection events as function of $^{235}$U concentration. Again, the pure gamma coincidences are independent of the $^{235}$U component in contrast to the neutron coincidences. The statistics on the pure neutron triple coincidences is relatively poor (d).

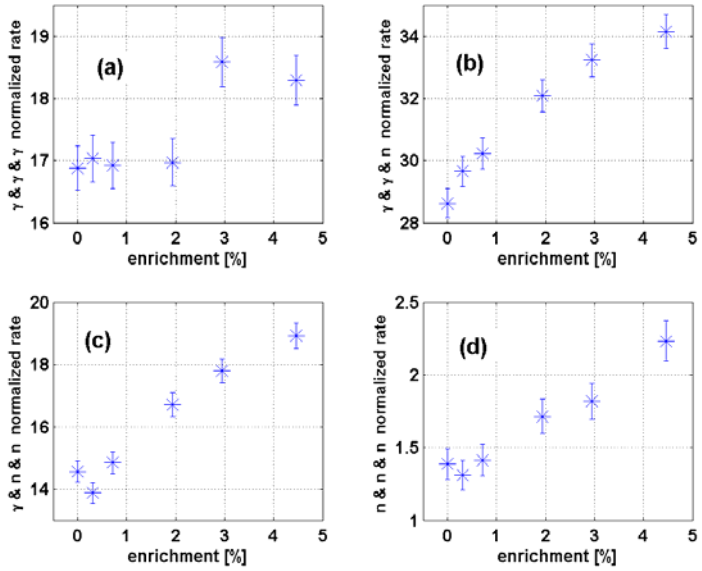



**Figure 6:** Thermal neutrons interrogations: triple coincidences for different kinds: γ-γ-γ (a), γ-γ-n (b), γ-n-n(c) and n-n-n (d) for the CBNM uranium standards and the empty CBNM container**.**

With the purpose of SNM detection of under practical conditions in mind, it is particularly desirable to detect the prompt fission neutron coincidences (as in Figure 5 and 6), not by means of thermal neutrons, but by epi-thermal neutron interrogation. The reason is the ability of non-thermal neutrons to penetrate practically any kind of shielding materials. The interrogating neutron flux however should be of low enough energy not to perturb the detection of fast fission neutrons in the PSD analysis. For this purpose the interrogating neutron spectrum achieved 55 μs after the 14-MeV burst (Figure 2) is useful. Figure 7 shows results of the epi-thermal neutron interrogation of the same set of CBNM samples as for the thermal interrogation.

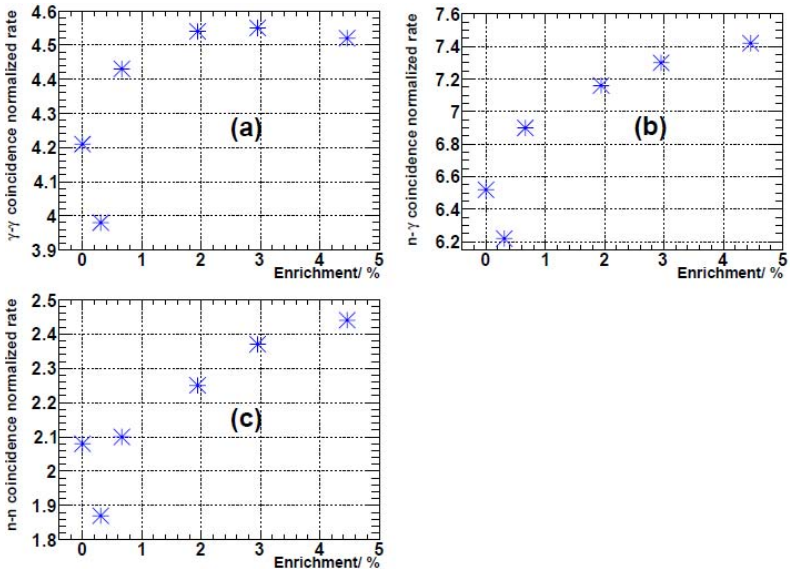



**Figure 7:** Epi-thermal neutron interrogation: double coincidences of different kinds: γ–γ (a), γ–n (b) and n-n(c) for the CBNM uranium standards and the empty CBNM container**.**

The results show the same trend as in Figure 5 with the exception that the depleted uranium sample (0.31% enrichment) gives a neutron coincidence response even lower than the empty container. A plausible explanation for this effect is perhaps found in the additional complications of epi-thermal interrogation, which is not the case for thermal interrogation, namely the competing neutron capture cross-section in addition to the induced fission cross-section in the resonance region. Figure 8 shows the behaviour of the neutron capture cross-section for the only elements in the CBNM samples: $^{235}$U, $^{238}$U and aluminium, as well as the fission cross-section for the two U isotopes. The interrogating epi-thermal spectrum of Figure 2 is also indicated. Aluminium appears not to play any role in the interrogation of the CBNM samples. The resonance capture in both $^{235}$U and $^{238}$U seem to play a role in reducing the neutron flux available for inducing fission in the $^{235}$U. Fission is $^{238}$U is excluded during the epi-thermal interrogation. Capture in $^{238}$U may well be the reason for the reduced fission rate in mainly the CBNM samples of low $^{235}$U concentration.

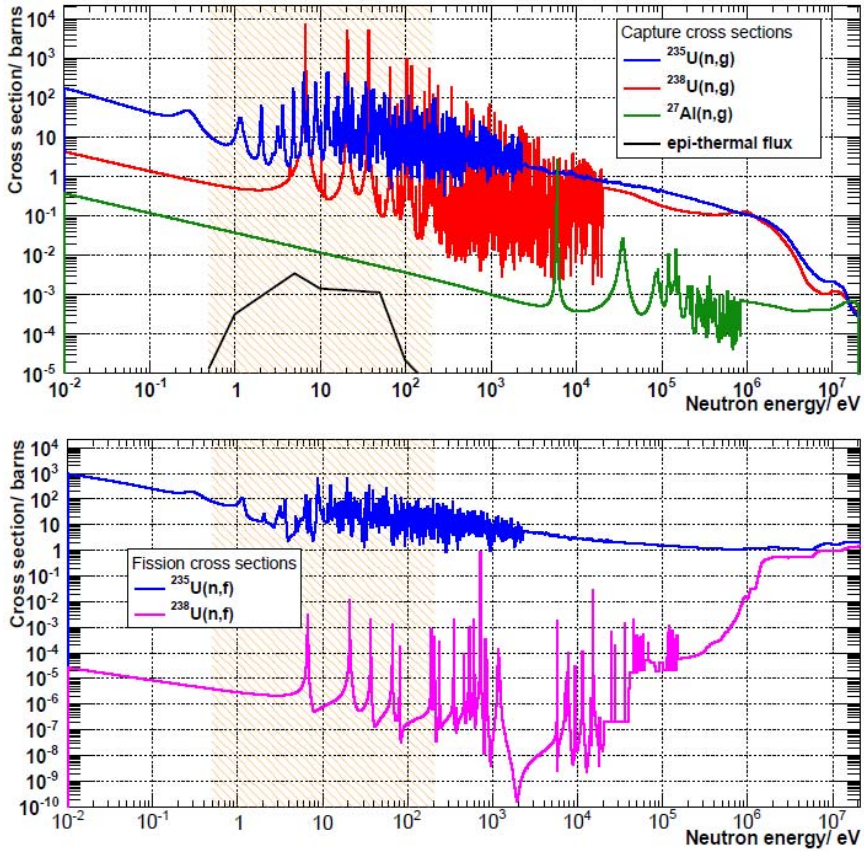
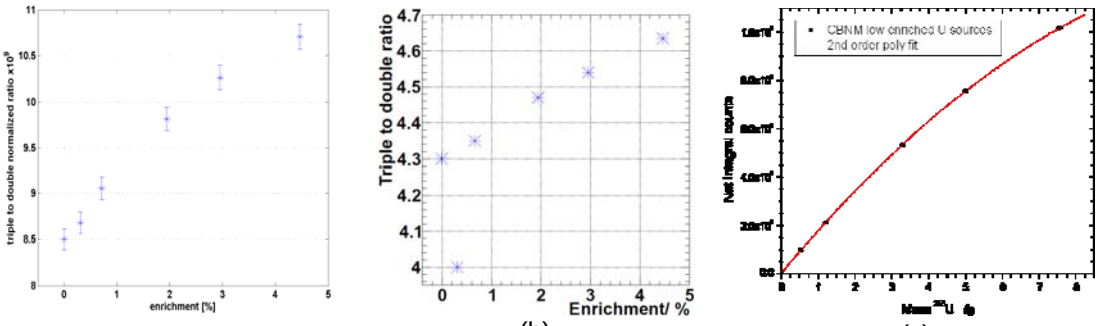



**Figure 8:** Capture cross-sections for $^{235}$U, $^{238}$U and aluminium (top), and fission cross-section for $^{235}$U and $^{238}$U (bottom) as function of neutron energy. In both picture the energy range of the PUNITA epi-thermal neutron flux is highlighted.

The triple coincidence response to epi-thermal interrogation has a similar trend as for the thermal interrogation depicted in Figure 6. But also in this case the capture reaction seems to dominate fission for low $^{235}$U concentrations.

When considering only the neutron response in the detectors (detection rate proportional to the fission rate), the ratio of triple to double coincidences may have a special significance. The triple to double coincidence ratio (TDCR) has been found to be proportional to the sample activity in scintillation detection systems for pure beta emission radionuclides. Ref 15 presents a review of

the TDCR method. This ratio is plotted in Figure 9 for the measurements above. The thermal interrogation (a) shows a behaviour which in fact reflects the fission rate in the sample. The epi-thermal interrogation likewise, but with the exception for small $^{235}$U concentrations as discussed above. For comparison thermal neutron interrogation of the CBNM samples with subsequent detection of prompt fission neutrons in the large array of $^{3}$He detectors (Figure 1) was performed. This method, known as the Differential Die-Away (DDA) technique, is very sensitive for small fissile quantities. The result, shown in Figure 9 (c), as function of $^{235}$U mass shows the same departure from linearity (due to thermal neutron self-attenuation) for the larger masses as is observed in the triple to double coincidence ratio.




(a) (b) (c)

**Figure 9:** Thermal (a) and epi-thermal (b) neutron interrogations: triple to double coincidence ratio normalized to the value of source monitor for CBNM uranium standards. The result of the DDA technique on the same samples is shown for comparison (c).

## 4. Conclusions

We have investigated the use of neutron interrogation combined with the detection of neutron coincidence events from induced fission as a method for detection of SNM. As an active method this concerns all kinds of SNM containing fissile isotopes. The detection system consists of multiple liquid scintillation detectors which allows to identify fast neutron interactions by means of the PSD analysis. Due to the fast response of these detectors, very short coincidence gates can be applied for the detection of the coincident prompt fission neutrons. These coincidence events are evidence of fission and consequently proof of the presence of fissile material in the object under investigation. A strong point of this method is the fact that the interrogating neutron flux does not affect the response from the sample in the detection system (when considering neutron coincidence events).

In the present work we investigated the practicality of this detection method by using three detectors in the PUNITA pulsed neutron facility. Results from interrogation with thermal and epi-thermal neutrons were compared. By assaying a series of CBNM standards with variable $^{235}$U concentration the neutron coincidence response showed a dependence on the fissile mass which is similar to that achieved by the high efficiency $^{3}$He based neutron detection system. Although the data achieved so far are preliminary, this fact gives confidence that the method can work even in an up-scaled configuration.

More experimental work is needed to provide confidence in the method and to determine actual detection limits, to improve the experimental setup (e.g. by applying a higher number of scintillation detectors), and to improve the efficiency of the coincidence circuitry.

The interrogation by epi-thermal neutrons is interesting for practical applications as this neutron source can penetrate a larger variety of shielding attempts. The epi-thermal interrogation however will not have as good a detection limit as the thermal interrogation due to the higher ratio

of capture to fission reactions in the epi-thermal range. More work is needed to quantify these competing effects both for bare samples and for samples embedded in matrix materials.

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# Applicability of Nonproliferation Tools and Concepts to Future Arms Control

**Mona Dreicer[1], Gotthard Stein[2]**

[1] Lawrence Livermore National Laboratory
Livermore, California, United States
[2] Consultant, Forschungszentrum Jülich GmbH
in der Helmholtz-Gemeinschaft 52425 Jülich, Germany
E-mail: dreicer1@llnl.gov, g.stein@fz-juelich.de

***Abstract:***

*A Working Group on Broader Perspectives on Nonproliferation and Nuclear Verification (WG3) was organized by the ESARDA/INMM International Safeguards and Nonproliferation and Arms Control Technical Divisions, in October 2011. The group considered how nonproliferation tools and culture could facilitate verification of future nuclear treaties. Two of the key challenges identified were providing confidence by monitoring and verification of the warhead lifecycle and fissile materials in States with nuclear weapons. These issues are complicated by a lack of consensus on disarmament goals, complexity of the weapons complex in some P-5 States, the technical capacity in countries without nuclear weapons, restrictions on the sharing of sensitive information to prevent proliferation, and the level of confidence in verifying compliance that can be achieved. Following-on from this discussion, a state-level approach could offer approaches to overcoming the obstacles that exist for verifying possible future arms control agreements. Next steps and potential R&D for technical verification and analysis are outlined.*

**Key Words: nonproliferation; arms control; verification**

## 1. Introduction

A Working Group on *Broader Perspectives on Nonproliferation and Nuclear Verification* was organized by the *INMM International Safeguards and Nonproliferation and Arms Control Technical Divisions* in the frame of the ESARDA/INMM conference Aix en Provence October 2011. The presentations focused on the technical topics related to international security and stability in global nonproliferation and arms control regimes, specifically asking how nonproliferation tools and culture might facilitate in the verification of future nuclear treaties with a focus on:

- Identifying existing tools and considering their applicability to the new challenges of verifying nuclear arms reductions?
- Modifications needed for this new context?
- Opportunities for use
- Existing gaps; and
- Needed R&D

The Working Group concluded[1] that a more systematic analysis of applying existing nonproliferation mechanisms might be useful, particularly for implementing international safeguards in weapons states, furthering the concept of remote monitoring, better understanding the implications of uncertainty in verification regimes, and the utility of applying a state-level approach, as currently being explored by the IAEA, to arms control. The group asked the following questions:

- "What is zero?" What will be accepted as "complete disarmament"?
- What would be an overarching framework for a network of verification regimes?
- Would verification standards change as reductions are implemented?
- Who are the Stakeholders: multilateral vs. bilateral parties, NWS vs. NNWS, or open society stakeholders such as NGOs, Industry, and the general public
- How can weapons-sensitive, proprietary, and classified information be protected?

Some of the challenges in verifying future arms control agreements might be addressed by various models for governance taking into account the questions highlighted above. Rather than aiming for a universal agreement, a step-by-step approach working to implement a web of complementary verification regimes is the most likely. This paper will not address the political aspects of this issue but focus on establishing technical mechanisms to increase transparency and verification. To build confidence in the ability to monitor and verify compliance, further development of the following concepts are needed: R&D for advancing technical capabilities, greater engagement with a wide variety of potential stakeholders (e.g P-5, IAEA, NNWS, United Nations, etc), and demonstration projects are needed. The ideas in this paper will be presented to the INMM Nonproliferation and Arms Control Division and the ESARDA Subgroup on Disarmament of the Novel Technology Subgroup and the VTM Group.

## 2. The Complexity

A weapons program is an interlinked complex of facilities and processes where materials (i.e. plutonium and/or uranium), components and weapons are produced, transported and stored (Figure 1). In some States, there are stages of the weapons lifecycle where civilian and military weapons activities are not clearly separated. There are various bilateral and multilateral programs that have worked to establish materials control, accounting and safeguards systems in nuclear weapons and non-nuclear weapons states. Although not simple to achieve, comprehensive safeguards systems could be put into place in State with nuclear weapons taking into account the protection of national security and proliferation-sensitive information[2].

The difficulty of designing a comprehensive warhead verification regime that could be verified with high-level of confidence has been debated since the 1960's (in the U.S. at least). If we assume that reductions of nuclear stockpiles will be accomplished by a network of different initiatives, a compatible set of continuity of knowledge regimes must begin by verifying baseline declarations and continue by monitoring movements of accountable items, any new production or dismantlement, transportation and storage for strategic nonstrategic, deployed and stored warheads. It will also be necessary to monitor irreversible dismantlement of warheads and safeguarding of resulting material, as well as new civilian nuclear material production not currently safeguarded under IAEA safeguards in nuclear weapons states of those not party to the NPT.

A framework will be needed to make progress in such a complex political, security and technical situation. Given a common approach, it might be possible to take information and inspections from the network of different agreements and initiatives and draw conclusions regarding a State's compliance with arms control commitments. Taking stock and learning from existing initiatives is the first step.

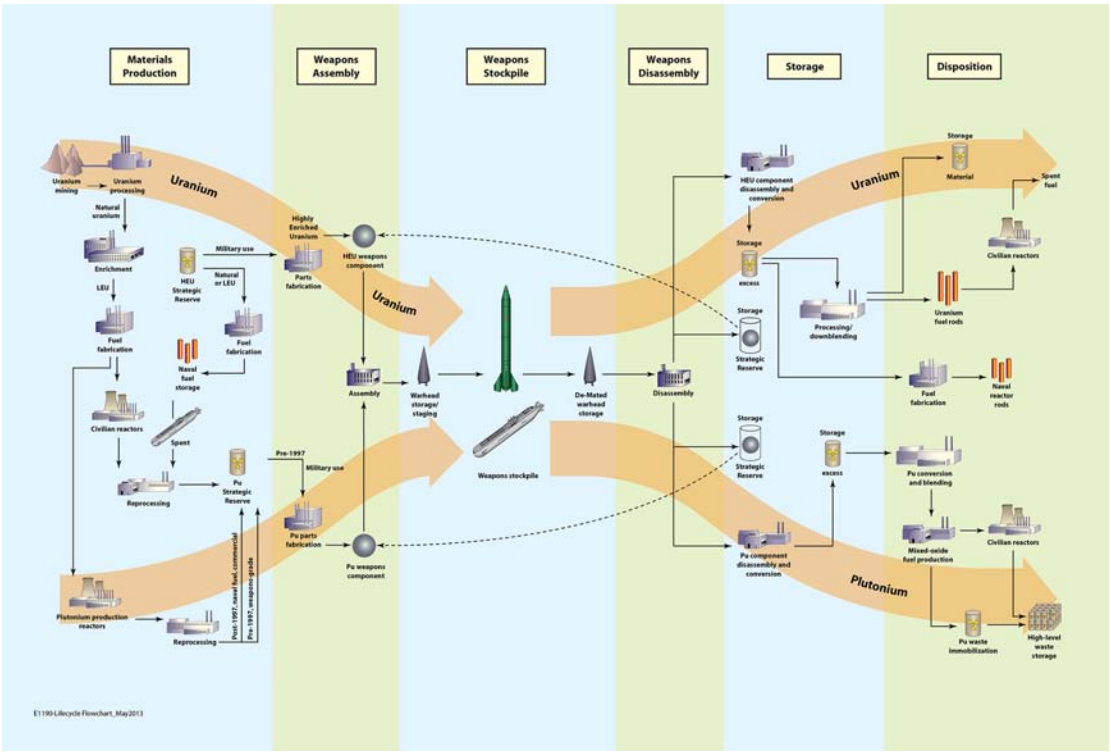


Figure 1. Nuclear materials and weapons lifecycle

## 3. Existing Mechanisms

There are many international nonproliferation programs that focus on nuclear security and/or material accountancy. IAEA Safeguards agreements and the Additional Protocol are the most prominent mechanism for verifying the peaceful uses of nuclear material in Non-Nuclear Weapons States. Negotiations to create some sort of fissile material control regime that would limit production of nuclear material for weapons in Nuclear Weapons States and States outside the Nuclear Nonproliferation Treaty has not yet been achieved but continues to be on the international security agenda. Initiatives undertaken bilaterally (U.S-Russia) and multilaterally (G8 Global Partnership) have worked to improve material accountancy in the Former Soviet Union and beyond. In addition, counterterrorism initiatives, not specifically nonproliferation-oriented, provide additional levels of assurances of responsible behavior or nonproliferation *bona fides*.

The U.S. and Russia have made some limited in-roads towards verifying parts of the nuclear weapons lifecycle. For example, transparency monitoring provisions under the 1993 United States-Russian Federation HEU Purchase Agreement allow for monitoring the down-blending of excess Russian weapons-origin highly enriched uranium to low enriched uranium; Plutonium Management and Disposition Agreement aims to dispose of excess weapons-grade plutonium monitored by inspectors, and the Plutonium Production Reactor Agreement allows for reciprocal monitoring of the 5 U.S. and 5 Russian shut-down plutonium production reactors. The START Treaty followed by New START, limits the numbers of strategic arms and provides verification and transparency via data exchanges/notifications and inspections/exhibitions in the U.S. and Russia. From 1996-2000, Russia, the U.S. and the IAEA worked to develop a system for verifying nuclear weapons disarmament and those who participated concluded that there were no technical obstacles to being able to implement such a regime (Trilateral Initiative).

## 4. Possibility of a State Level Approach

The IAEA has been improving its processes for safeguards implementation to ensure it is objectives-based and information-driven. The IAEA State-Level Concept (SLC)[3,4] outlines three phases:

1. Developing State-level safeguards approaches
2. Planning & conducting safeguards activities, and
3. Establishing knowledge & drawing conclusions

Decades of experience has taught the IAEA that integrating a wide variety of information in an objective way is the most effective way to analyze monitoring data and provide the information that States need to make their verification decisions.

To address the first phase, a methodology to implement a State-level Approach (SLA) allows State-specific approaches to nuclear safeguards, i.e. differentiation between States, taking into account all information available to the IAEA and being responsive to changes in risk assessment. It is not limited only to States with an Additional Protocol and it is applicable to all States with a safeguards agreement in force. This process is being implemented in three steps[5]:

- Identification of plausible acquisition paths,
- Specification and prioritization of State-specific technical objectives,
- Identification of safeguards measures to address the technical objectives.

The SLA outlines sequences of activities (acquisition paths, AP) that a State could consider to acquire weapons usable material. It analyzes all plausible APs aiming to determine whether a proposed set of safeguards measures will be sufficient. Mapping out the APs is essentially producing a state-specific network of process and material and flows with identified nodes for inter-connections[6]. The "relative attractiveness" of an AP, or usefulness in nuclear weapons, is considered in addition to the time it would take to implement such a process in a country. Currently, the IAEA is using this approach based on expert judgment, but a more transparent and reproducible method is needed[4].

A framework for collection and analysis of monitoring/verification information related to new arms control agreements could be formulated using the same characteristics being proposed for IAEA Safeguards[5]:

- objective – use the same method and criteria for all States;
- transparent - analysis should be open; understood methodologies;
- standardized - steps of the analysis have to be defined in detail;
- reproducible - conclusions not depending on who performs the analysis; and
- document - decisions made during the analysis have to be protocoled.

SLC Phases two and three would depend very much on the scope of the new agreements addressing the different parts of the nuclear weapons lifecycle. A common methodology cross the new agreements/initiatives could facilitate state-level analysis and conclusions. Table 1 presents steps that could be taken to developing such an approach and aid in design of a future program of research.

| Phases | | Focus & Questions |
|---|---|---|
| Develop approach | Analyze cheating pathways and level of risk | Taking into account governance, technology, cooperative measures, national technical means |
| | Monitoring & Verification Measures/technologies | Do stakeholders consider methods sufficient? |
| | Acceptable level of uncertainty | Bi-lateral, regional and/or global |
| Planning and conducting Activities | Existing treaties, agreements and regimes | Gaps in the network |
| | Identification of Verification Gaps | Existing technology |
| | Design new initiatives | Political feasibility |
| Establishing knowledge | Collection and analysis of data | Dealing with sensitive, classified and proprietary information |
| | Use of variety of data sources: NTM, unilateral, treaty-based and open source information | Managing types of data from different countries and regions in different time scales; data volumes |
| | | Consultation & Clarification processes |
| Drawing conclusions | Influence of trust on data interpretation & objectivity | Lessons learned from the IAEA? |
| | Critical review/adjudication of varying interpretations | |
| | Presentation of data to analysts and decision-makers | |

Table 1. Considerations on developing a state-level approach for achieving confidence, for possible future arms reductions.

## 5. Research and Development for Verification Technologies

Whatever governance and analytical framework is constructed, verification technologies are going to be needed for implementing any future regime(s). The R&D conducted to support IAEA safeguards and counterterrorism initiatives concentrate on materials detection and accounting. Verification mechanisms needed to monitor and verify the phases of the weapons lifecycle are at an earlier stage of development. Table 2 provides a list of possible research, development and deployment needs for possible cooperative measures that might be used in future verification initiatives.

## 6. Summary

It does not seem likely that a global arms reductions agreement could be achieved. It is more practical to work on a step-by-step approach to implement a web of complementary regimes that might ultimately achieve this goal. When the political challenges are overcome, technical and analytical capabilities will be needed to maintain confidence via increased transparency and technical verification. To build confidence in the ability to monitor and verify compliance, a consistent analytical framework, R&D advances, and demonstration projects will be needed. We have proposed a state-level analytical approach be considered in developing future arms reductions initiatives, which has been based on the IAEA's State-Level Approach, and technical R&D needed to support implementation.

| Monitoring Objective | Possible Cooperative Measures | Research, Development & Deployment Needs |
|---|---|---|
| Non-deployed<br>Monitored storage | Counting/Transparency<br>Declarations<br>Continuity of knowledge<br>Routine inspections<br>Remote monitoring | RDE – confirmation of warhead contained in container (attributes, imaging) (HEU, Pu)<br>Information barriers/Managed Access; authentication<br>Tags/unique identifiers for containers<br>Remote monitoring |
| • Production<br>• Weaponization | • Multiple steps with increasing intrusiveness<br>• Declarations and Accountancy<br>• Routine to short notice inspections<br>• Continuity of knowledge | • Tags for warheads & components<br>• Information barriers<br>• Managed Access<br>• Satellite Imagery<br>• Environmental Monitoring<br>• Forensics |
| • Dismantlement | • Done in multiple steps with increasing intrusiveness<br>• Declarations<br>• Transparency to counting to accounting<br>• Continuity of knowledge<br>• Routine to short notice inspections | • Tagging components & materials (to track from dismantled components fissile material)<br>• Managed Access<br>• Satellite Imagery<br>• Warhead measurements using RDE |
| • State-level | • Cradle-to-grave tracking (State-level continuity of knowledge)<br>• Inspections | • Information analysis<br>  • Sampling Statistics<br>  • Propagation of Uncertainty<br>  • Game theory<br>• Societal Monitoring |

Table 2. Proposal for the development of verification technologies

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# Proliferation Resistance Considerations within the Collaborative Project for a European Sodium Fast Reactor

**Fatih Alim, Giacomo G.M. Cojazzi, Guido Renda**

Institute for Transuranium Elements (ITU)
Joint Research Centre, European Commission
Via Fermi n. 2749, Ispra 21027 (VA) Italy
E-mail: fatih.alim@jrc.ec.europa.eu

***Abstract:***

*The collaborative project for a European Sodium Fast Reactor (CP-ESFR) is an international project where 25 European partners developed R&D solutions and concepts for a European sodium fast reactor. The project was funded by the 7th EU Framework Programme and covered topics such that the fuel, the fuel element and the fuel cycle, the safety concepts, the reactor architectures and components and the balance of plant. Within sub-project 3, dedicated to safety, a task addressed proliferation resistance considerations. The Generation IV International Forum (GIF) Proliferation Resistance & Physical Protection (PR&PP) Evaluation Methodology has been selected as the general framework for this work, complemented by punctual aspects of IAEA-INPRO Proliferation Resistance (PR) assessment methodology and other literature studies in particular for material type characterization. In the paper the main core characteristics and the so called working horses, for a loop and a pool Sodium fast reactor concept are firstly presented, by highlighting those features more relevant for the proliferation resistance. Selected activities carried out in the project for its proliferation resistance considerations are then illustrated.*

**Keywords:** proliferation resistance; sodium fast reactors; PR&PP.

## 1. Introduction

The collaborative project on the sodium fast reactor (CP-ESFR) [1] is an international project where 25 European partners develop R&D solutions for a European Sodium Fast Reactor concept [2, 3]. The Project explores aspects related to the main design aspects of the system, such as the fuel, the fuel element and the fuel cycle, the safety concepts, the reactor architectures and its components and the balance of plant. Within sub-project 3, dedicated to the system's safety concepts, a specific task led by JRC-ITU, with contributions of AREVA, EdF and ENEA addressed proliferation resistance issues [4].

The aim of this task was to propose an approach to evaluate the level of protection of a given NPP design with respect to nuclear proliferation. Before doing applied studies, an approach was to be defined to take into account the methods and the parameters used to evaluate the proliferation resistance aspects of interest. This had to integrate the contributions of the Generation IV International Forum (GIF) Proliferation Resistance & Physical Protection working group (PR&PP) activities [5, 6], as well as the activities on proliferation resistance (PR) carried out within the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) of IAEA [7]. In order to collect the system's PR relevant information, the structure of the GIF PR&PP White papers [8] has been chosen, and following it a reflection on the system's possible response to the GIF proliferation threats (concealed diversion, concealed misuse, breakout and replication of technology on clandestine sites) has been performed in a qualitative way.

Since a full PR analysis for the entire system was beyond the scope of this task, the high level reflections made following the GIF white paper helped in identifying the diversion scenario as the most interesting one to investigate further. The focus was therefore put on a more detailed analysis of the

impact of the core alternatives under consideration on a potential diversion scenario. The study identified the potential diversion targets, the potential diversion points for each target and performed an analysis of the attractiveness of the available targets for a potential proliferant State. This activity has been carried out taking the GIF Proliferation Resistance and Physical Protection Evaluation Methodology (GIF PR&PP EM) [5] and its Addendum [6] as the general guideline for identifying targets and potential diversion points. The targets proliferation attractiveness has been analyzed in terms of the suitability of the targets' nuclear material as the basis for a weaponization stage. Since the PR&PP Fissile Material Type measure was felt too coarse for a thorough investigation, other literature studies and their related metrics have been applied [7, 9, 10, 11] to the nuclear material items present in the considered core alternatives [4, 12].

A nuclear system's proliferation resistance emerges from the synergy of intrinsic features and institutional measures. To provide some guidance to ESFR designers in terms of features that would facilitate the application of international safeguards to their system, a high-level safeguardability analysis was performed, and two possible illustrative safeguards approaches for the ESFR system were proposed, one based on near real time monitoring of the nuclear inventory and one based on current practice in European water reactors.

This paper will summarize the main ESFR design aspects relevant for PR and illustrate the high level reflections made following the GIF white paper by presenting the structure of the GIF PR&PP White Paper template and its application to the ESFR reactor.

## 2. The GIF PR&PP White Paper Template on six Generation IV systems

In 2011 The Generation IV Proliferation Resistance and Physical Protection Evaluation Methodology Working Group and the six System Steering Committees of the Generation IV International Forum published a joint report [8] illustrating the main PR and PP features of the six Generation IV conceptual designs (gas-cooled fast reactor – GFR -, very-high-temperature reactor – VHTR -, supercritical-water-cooled reactor – SCWR -, sodium-cooled fast reactor – SFR -, lead-cooled fast reactor – LFR -, molten salt reactor – MSR -), together with a number of common cross-cutting issues.

The report was the outcome of a joint effort of the above-mentioned groups, implying a close collaboration via workshops and the compilation of a PR&PP White Paper template for each of the six designs. The structure of the White Paper template aims to capture the relevant pieces of information needed for a PR&PP analysis – PR&PP relevant design description and PR&PP considerations - in a structured way, and foresees the following chapters [8]:

1. Overview of Technology
2. Overview of Fuel Cycle(s)
3. PR&PP Relevant System Elements and Potential Adversary Targets
4. Proliferation Resistance Considerations Incorporated into Design
5. Physical Protection Considerations Incorporated into Design

For the compilation of chapters 3 (PR&PP Relevant System Elements and Potential Adversary Targets), 4 (Proliferation Resistance Considerations Incorporated into Design) and 5 (Physical Protection Considerations Incorporated into Design), the activity foresees the application of the PR&PP Evaluation Methodology framework, illustrated in Figure 1.

The Generation IV International Forum defines proliferation resistance as "*that characteristic of an NES that impedes the diversion or undeclared production of nuclear material or misuse of technology by the Host State seeking to acquire nuclear weapons or other nuclear explosive devices*" [5]. According to the PR&PP methodology the proliferation threats to be considered for a host state actor are 1) *Concealed diversion,* 2) *Concealed production of nuclear material,* 3) *Breakout* and 4) *Production in clandestine facilities*. Threats of non-state actors are addressed in the context of physical protection and will not be further considered here.

A full PR evaluation of the systems design was beyond the scope of the project. The GIF PR&PP methodology framework has been used to decompose the system into system elements and to

identify and characterize potential diversion targets. The analysis then focused on high-level considerations on the four above-mentioned proliferation threats.

Given the outcome of the compilation of the ESFR White Paper draft, the activity deepened the analysis on the aspects that were considered to be more relevant for the project, i.e. targets characterizations in terms of nuclear material attractiveness for a proliferator and safeguardability considerations[1]. Limited diversion analysis were also carried out to identify for the different diversion targets, possible diversion points, related strategies, and an high level description of the pathways and of the proliferation actions.[1].

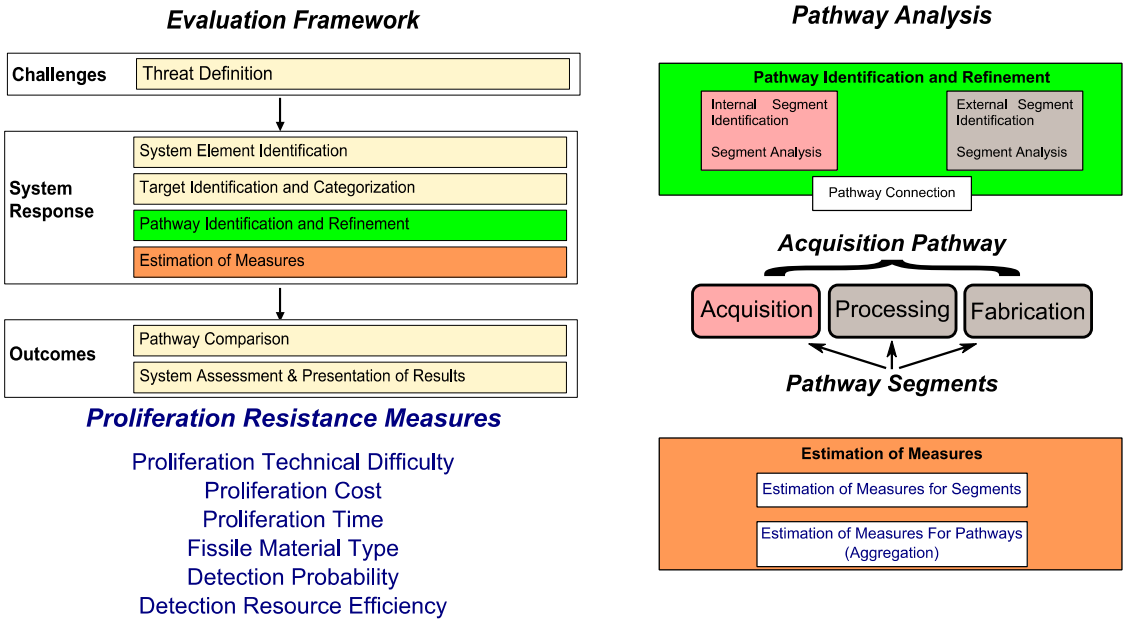


**Figure 1:** The GIF PR&PP methodology framework [5].

The following sections will illustrate the analysis carried out and therefore follows the above-described structure.

## 3. Aspects of the ESFR design relevant to proliferation resistance (Overview of Technology)

Two concepts of 1500 MWe reactors called “Working Horses” have been proposed in the context of CP ESFR: a pool and a loop design. For both designs, two core options have been considered: one with U and Pu oxide fuel and the other with U and Pu carbide fuel. For both carbide and oxide cores, the inner and outer fuel regions have different Pu mass content. The reactor architectures and the core of the working horses are shown in Figure 2, for the oxide fuel option.

A comparison of the working horses’ characteristics with respect to the sodium fast reactor designs considered in the GIF sodium fast reactor white paper [8] is shown in Table 1.

[1] These aspects, covered by the task activities, will not be presented in this paper.

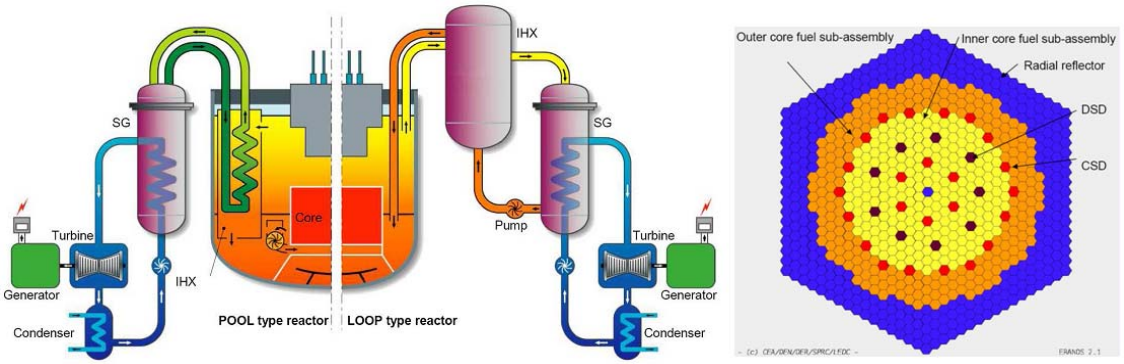


**Figure 2:** Schematic diagram of reactor architectures of the two ESFR working horses [13] and the SFR core with oxide fuel, no blanket case. CFD stands for Control and Shutdown device; DSD stands for Diverse Shutdown System. [14]

The systems could be used for minor actinide (MA) management, either in homogeneous or heterogeneous mode. Homogeneous MA management basically consists in replacing part of the uranium in the fresh fuel elements with minor actinides (4% MA), while heterogeneous MA management considers an additional radial blanket ring of fuel assemblies (FA) for the specified 'working horse' core [6].

Fresh radial blanket assemblies, when present, are always composed of depleted $UO_2$ (80%) and MAs (20%). The project partners investigated the possibility to perform MA management only within the oxide core.

Table 2 presents the main characteristics of the CP-ESFR fresh fuel elements, Table 3 reports the plutonium vector of the fresh fuel assemblies when homogeneous MA transmutation is considered, and Table 4 illustrates the related MA composition. The fresh fuel assemblies contain reactor-grade plutonium coming from spent LWR reprocessed fuel.

| | SFR Design Concepts in [8] | | | ESFR | |
|---|---|---|---|---|---|
| **Design Parameters** | **JSFR[2]** | **KALIMER[3]** | **SMFR[4]** | **Oxide** | **Carbide** |
| **Power Rating, MWe** | 1,500 | 600 | 50 | 1500 | |
| **Thermal Power, MWt** | 3,570 | 1,525 | 125 | 3600 | |
| **Plant Efficiency, %** | 42 | 42 | ~38 | 42 | |
| **Core outlet coolant temperature, °C** | 550 | 545 | ~510 | 545 | |
| **Core inlet coolant temperature, °C** | 395 | 370 | ~355 | 395 | |
| **Main steam temperature, °C** | 503 | 495 | 480 | 470 | |
| **Main steam Pressure, MPa** | 16.7 | 16.5 | 20 | 18.5 | |
| **Cycle length, years** | 1.5–2.2 | 1.5 | 30 | 2050 EFPD[5] | 1600 EFPD |
| **Fuel reload batch, batches** | 4 | 4 | 1 | 5 | 3 |
| **Core Diameter, m** | 5.1 | 3.5 | 1.75 | 4.72 | 4.10 |
| **Core Height, m** | 1.0 | 0.8 | 1.0 | 1.0 | 0.8 |
| **Fuel Type** | MOX(TRU bearing) | Metal(U-TRU-10%Zr Alloy), | Metal(U-TRU-10%Zr Alloy), | $(U,Pu)O_2$ | (U,Pu)C |
| **Pu enrichment (Pu/HM), %** | 13.8 | 24.9 | 15.0 | 14.05-16.35 | 17.80-24.50 |
| **Burn-up, GWd/t** | 150 | 79 | ~87 | 100 | 144 |
| **Breeding ratio** | 1.0–1.2 | 1.0 | 1.0 | - | - |

**Table 1:** CP-ESFR reference cores characteristics compared with the SFR systems mentioned in [8].

[2] JAEA Sodium Fast Reactor. The system foresees a loop design [8].
[3] Korean pool-type SFR design [8].
[4] US pool-type Small Modular SFR design [8].
[5] Effective Full Power Days.

| | Oxide Core | Carbide Core |
|---|---|---|
| **Pu mass content (inner fuel region) (%)** | 14.05 | 17.8 |
| **Pu mass content (outer fuel region) (%)** | 16.35 | 24.5 |
| **Pu mass per item** | > 1SQ[6] | > 1SQ |

**Table 2:** Main characteristics of the CP-ESFR reference cores fresh fuel assemblies [4].

| **Pu Isotope** | **$^{238}$Pu** | **$^{239}$Pu** | **$^{240}$Pu** | **$^{241}$Pu** | **$^{242}$Pu** |
|---|---|---|---|---|---|
| **% w** | 3.60 | 47.76 | 29.89 | 8.29 | 10.46 |

**Table 3:** Pu vector for CP-ESFR reference cores fresh fuel assemblies in case of homogeneous MA transmutation [15].

| **Isotope** | **$^{237}$Np** | **$^{241}$Am** | **$^{242m}$Am** | **$^{243}$Am** | **$^{242}$Cm** | **$^{243}$Cm** | **$^{244}$Cm** | **$^{245}$Cm** | **$^{246}$Cm** |
|---|---|---|---|---|---|---|---|---|---|
| **% w** | 16.86 | 60.62 | 0.24 | 15.70 | 0.02 | 0.07 | 5.14 | 1.26 | 0.09 |

**Table 4:** MA composition of the CP-ESFR reference cores fresh fuel assemblies in case of homogeneous MA transmutation [15].

The original core options ("reference cores") have subsequently been subject to further analysis and new variants were put on the table for consideration ("optimized cores"). As long as PR is considered, the major design variation of the optimized cores is the availability of an axial blanket beneath the active part of the fuel element.

The main difference between the optimized and reference cores characteristics are reported in Table 5.

| | Oxide Core | | Carbide Core | |
|---|---|---|---|---|
| | **Reference Core** | **Optimised Core** | **Reference Core** | **Optimized Core** |
| **Number of Inner core S/A** | 225 | 225 | 168 | 225 |
| **Number of Outer Core S/A** | 228 | 228 | 246 | 228 |
| **Active Core Height (m)** | 1.0 | 1.0 | 0.8 | 1.0 |
| **Number of fuel pins per sub-assembly** | 271 | 271 | 331 | 271 |
| **Outer clad diameter (mm)** | 10.73 | 10.73 | 8.0 | 8.5 |
| **Inner clad diameter (mm)** | 9.73 | 9.73 | 7.0 | 7.5 |
| **Fuel pellet diameter (mm)** | 9.43 | 9.43 | 6.87 | 7.37 |
| **Fuel pellet material** | $(U, Pu)O_2$ | $(U, Pu)O_2$ | (U, Pu)C | (U, Pu)C |
| **Inner Core Fuel Enrichment (atom%)** | 14.05 | 14.76 | 17.8 | 14.05 |
| **Outer Core Fuel Enrichment (atom%)** | 16.35 | 17.15 | 24.5 | 18.35 |

**Table 5:** Comparison of reference and optimized cores characteristics [17].

[6] The IAEA defines the significant quantity (SQ) as the approximate amount of NM for which the possibility of manufacturing a nuclear explosive device cannot be excluded. A SQ for Pu, containing less than 80% $^{238}$Pu, is 8 kg [16].

## 4. ESFR PR relevant system elements and potential adversary targets

As defined in [5], "*PR relevant system elements"* are defined as a *"collection of facilities inside the identified nuclear energy system (NES) where nuclear material diversion/acquisition and/or processing, as well as theft or radiological sabotage could take place*", and proliferation targets are "*nuclear material that can be diverted or equipment/processes that can be misused to process undeclared nuclear materials or can be replicated in an undeclared facility*".

Figure 3 shows the basic system elements for both pool and loop type working horses[7]. Although the in vessel fuel handling systems of the reactors are different for pool and loop reactor types, both designs share the same conceptual system elements, shown in Figure 3.

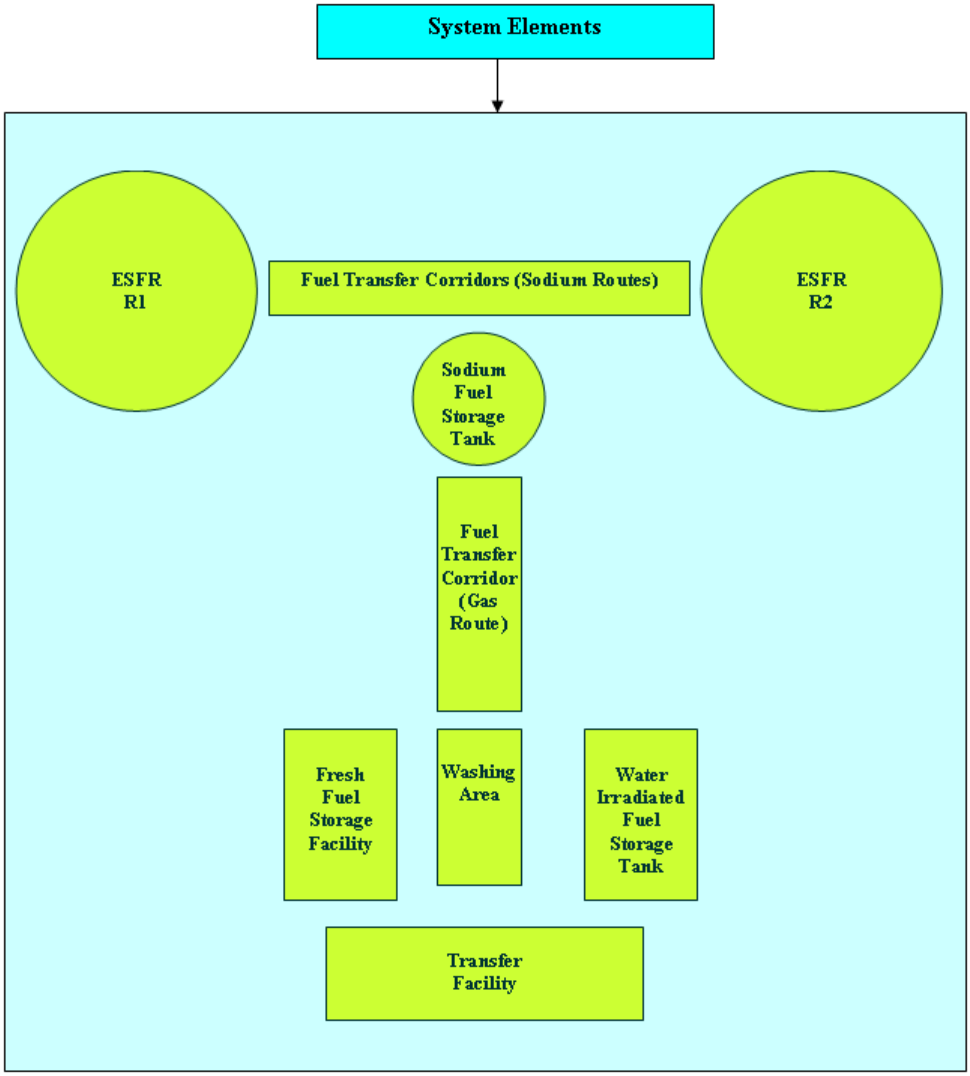


**Figure 3:** ESFR system elements.

[7] In case MA-bearing fuel is considered, there might be the need of adding a disassembly shop after the washing area or in the transfer facility allowing transport to the reprocessing plant. This would be required also for assembling the FAs with MA. This would add an additional system elements to the ones presented in the figures [4].

### Targets for the reference cores

The possible ESFR diversion targets for the reference cores are the Fresh FAs, spent FAs and - if present – radial blanket assemblies. Figure 4 shows the possible diversion target types (in red) in the cases simulated within the collaborative project. The considered ESFR fuel assemblies have a much higher fissile material content compared to typical light water reactors. This makes the ESFR fuel assemblies, especially the fresh ones, an attractive target for a potential diversion strategy.

Under normal conditions, all fresh FAs, all spent FAs and the spent (irradiated) radial blanket assemblies contain Pu of RG type[8]. The Pu quality in the fresh fuel is that of the associated fuel cycle.

An additional adversary target is represented by the undeclared nuclear material production (by misusing the ESFR system) via the irradiation of fertile material in the reactor core [8].

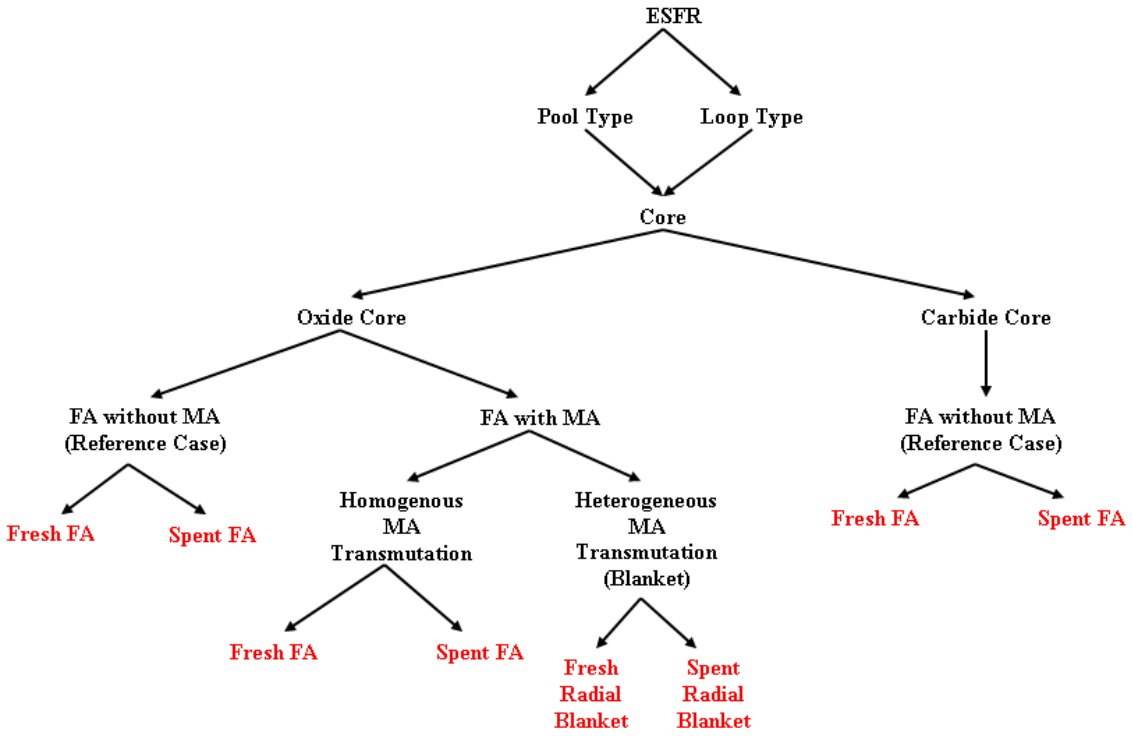


**Figure 4:** Reference cores diversion targets [4].

### Targets for the optimized cores

The identified diversion targets for the optimized cores are the fresh FAs, spent FAs, and radial blanket assemblies. The optimized cores foresee the presence of a lower axial blanket in fresh FA made of depleted uranium. When the blanket is irradiated, an attractive Pu quality is produced and after irradiation the lower axial blanket in spent FA can be cut from the rest of the assembly in order to be processed separately. For this reason, Figure 5 shows also the lower axial blanket in spent FA as an additional potential adversary target to the already mentioned ones[9].

[8] Here the categorization of PR&PP WG [5] for Pu is adopted. PR&PP distinguishes Pu in weapon grade (WG) Pu, reactor grade (RG) Pu and deep burn (DB) Pu on the basis of the percentage of fissile Pu isotopes present (WG, nominally 94 % fissile Pu isotopes, RG nominally 70% fissile Pu isotopes, DB nominally 43 % fissile Pu isotopes. More detailed material type studies have been carried out in the project by using different metrics.

[9] In order to divert the axial blanket material, the diverter has to divert the full ESFR spent fuel assemblies.

With the exception of lower axial blankets in spent FAs, all fresh FAs, all spent FAs and the spent (irradiated) radial blanket assemblies contain Pu of RG type. The lower axial blanket in spent FA contain weapon grade Pu. Fresh radial blankets of both reference and optimized cores are Pu free.

As with the reference cores, another potential adversary target is represented by the undeclared nuclear material production via the irradiation of fertile material in the reactor core [8].

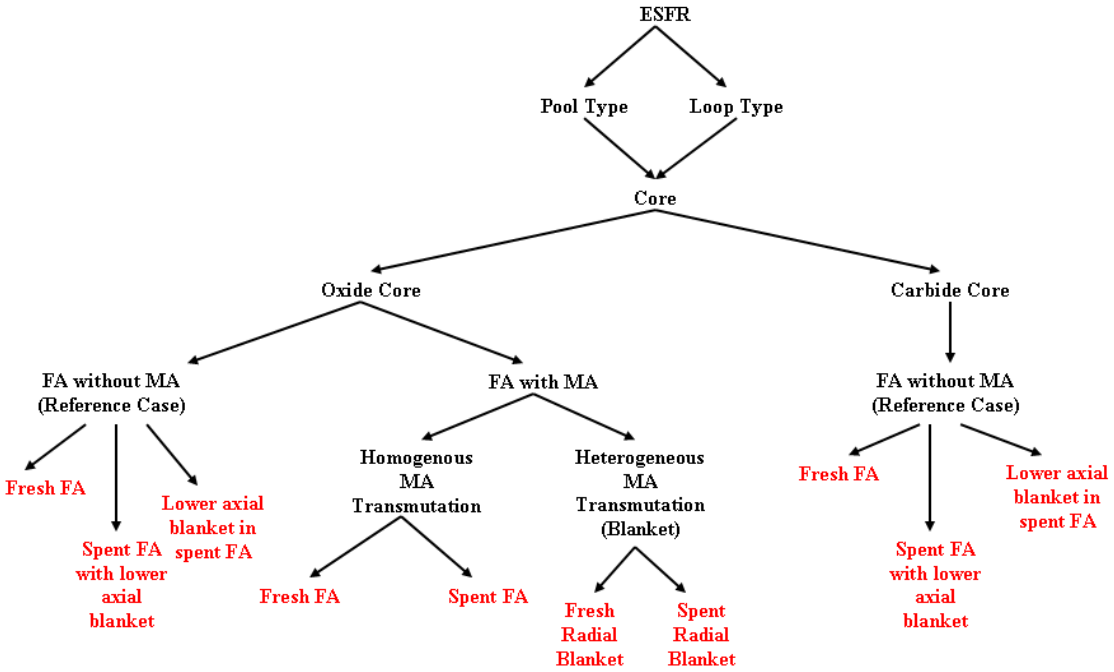


**Figure 5:** Optimized cores diversion targets [4]

## 5. ESFR proliferation resistance considerations

The PR strategies to be considered under a GIF PR analysis are the following [5]:

- Concealed diversion;
- Concealed production of nuclear material;
- Breakout;
- Production in clandestine facilities.

The following sections discuss the possible ESFR response to the considered proliferation strategies.

**Concealed diversion**

The oxide reference fresh core foresees a load of 11.6 tons of Pu and the carbide one foresees a load of 8.5 tons of Pu [18]. Each reference oxide core fresh fuel assembly contains an average of 25.6 kg of Pu and each carbide core's fresh fuel assembly contains an average of 20.5 kg of Pu[10], resulting respectively in 3.2 and 2.6 significant quantities (SQs) of Pu for each oxide and carbide core's assembly. Moreover the number of significant quantities in the fuel assembly does not change significantly with burn-up. This means that the diversion of one ESFR fuel assembly (either fresh or irradiated) is enough to obtain more than 1 SQ of plutonium. For a typical LWR spent fuel assembly, one fuel assembly contains less than 1 SQ of Pu. From this point of view, the ESFR fuel assemblies might represent a more attractive diversion target than those of a typical light water reactor, the quality of Pu being of reactor grade in both cases.

[10] These values are calculated using the data in [18]

Within ESFR, actinide management was explored for the case of the oxide fuel. As highlighted by [8], "*handling methods for fresh fuel assemblies may depend significantly on minor actinide content (homogeneous recycle or heterogeneous recycle concentrated minor actinide targets)*"[11]. MA bearing fresh fuel, as in the case of homogeneous minor actinide transmutation, has a higher radiation emission, making the handling of the fuel assemblies more complicate. This higher radiological barrier might increase proliferation resistance[12]. On the other hand, MA bearing fuel assemblies are more difficult to fabricate and might bring in new measurement challenges for the safeguards inspectors. Moreover, depending on MA-content, the ESFR system might need an on-site assembly shop which, despite radiation levels, may provide avenues for concealed introduction of fertile materials. This, combined with a more difficult irradiated FA composition analysis, would potentially enable concealed diversion and especially undeclared production of nuclear material, even though in fractional amounts of SQ per FA. Despite the presence of MA in fresh fuel, the resulting Pu in spent fuel will remain of reactor grade quality.

Spent blanket elements could have relatively low burn-up and could contain high quality plutonium material. In particular, radial blanket assemblies are normally sensitive items from a PR point of view. Using radial blanket in the design might cause a potential PR concern.

The addition of MA to the fertile radial blanket assemblies (heterogeneous minor actinides transmutation) might constitute a proliferation resistance feature and might provide additional technical difficulty to potential proliferators. Within ESFR, the doping of depleted uranium with 15-20 % of MA results – after irradiation - in Pu with a very high $^{238}$Pu content, thanks to the presence of Np in the MA composition. This will constitute an important proliferation barrier in terms of nuclear material attractiveness. According to the PR&PP categorization the Pu of spent blanket will be RG Pu.

Both ESFR oxide and carbide optimized core foresee the presence of a lower axial blanket in fresh fuel made of depleted uranium which, when irradiated, will result in weapon grade Pu, this however will be diluted and several FA will need to be diverted to obtain an SQ of Pu (8 FA for spent oxide fuel and 10 for the case of spent carbide fuel).

The sodium environment in many system elements represents a challenge to nuclear safeguards activities. Due to the sodium being visually opaque, new methods might be needed to identify and monitor fuel assemblies.

**Concealed production of nuclear material (misuse)**

There are multiple possibilities for misusing a system, and therefore there are many different possible pathway combinations. The weaker pathways with respect to PR should be identified and analyzed in detail and related safeguards requirements should be identified. It has to be noticed that generally a misuse strategy has the objective to produce a better-than-available fissile material quality (in this case weapon-grade plutonium).

Undeclared fissile material production by irradiating fertile material in inner, outer and blanket regions of the ESFR reference core might be a potentially attractive misuse strategy: replacing a FA in the inner or outer core region with a fertile target assembly could be a possible route for the concealed production of fissile nuclear material. The fertile material could be put in the fuel part of the FA or in the parts of the assembly just above or beneath the active part.

For the ESFR optimized cores, the plutonium available in the irradiated axial blanket is already weapon-grade, making the concealed production of better grade nuclear material not necessary and making concealed production a less interesting proliferation strategy then concealed diversion.

The other possible goal of a misuse strategy could be to produce a bigger quantity of high-quality nuclear material. The ESFR axial blanket is distributed over all the fresh fuel assemblies, resulting in having only a small part of the assembly containing weapon-grade plutonium. This would require the

[11] [8], Section 4.1.

[12] Although PR might be increased by the higher radiological barrier, its effectiveness in deterring a potential proliferator from diverting the target is debatable. For a discussion on this topic see [19]

diversion of several assemblies to acquire one significant quantity of weapon-grade plutonium. Trying to misuse the optimized reactor core to end up with a single element containing one or more significant quantities might represent a potentially attractive scenario. From this point of view, the ESFR system doesn't present peculiarities making it substantially different from other existing sodium fast reactors, and therefore its behavior in a misuse scenario would be similar to any other SFR.

In case of a concealed misuse scenario, the proliferator's biggest challenge is to avoid detection by the safeguards system. One of the proposed safeguards approaches within the CP-ESFR project activities relies on a near real-time monitoring of the reactor system, making this strategy particularly difficult to be implemented.

**Breakout**

Carbide and oxide reference fresh fuel assemblies contain on average 3.2 and 2.6 SQ of plutonium respectively. Thus, a breakout scenario would see the availability of a huge amount of significant quantities. According to IAEA, the conversion time for MOX fuel is of the order of 1-3 weeks [16]. However this material would be reactor grade Pu and not weapon-grade Pu, i.e. less than ideal for a weapon programme. The major concern posed by a breakout scenario would be the huge production capability of Pu of any desired quality, and the key parameter to assess would be proliferation time.

**Production in clandestine facilities**

As highlighted by other literature studies, the sodium fast reactor technology does not seem the most suitable one to be replicated in a clandestine facility. The intrinsic difficulties connected with the presence of sodium, together with the overkilling dimensions of a fully-fledged commercial power reactor makes the ESFR a very unlikely candidate for clandestine replication.

## 6. Conclusions

Following the GIF White Paper Structure, information has been collected for the two CP-ESFR working horses and related core concepts with emphasis on features relevant for PR. Both loop and pool working horses share the same core concepts. Although SFRs are item facilities and this facilitates accounting and control for safeguards purposes, liquid sodium is a hostile environment and an opaque coolant that makes the verification when the fuel element is under sodium very difficult. Both working horses foresee batch operation enabling the possibility to seal the core between reloading and reshuffling operations. In terms of safeguards it is of paramount importance to identify and track, possibly in real time, the location and – if feasible - the composition of the fuel elements.

From a PR point of view the two working horses options (pool or loop) are equivalent, provided that they are considered with the same core configuration, i.e. oxide (with or without MA) or carbide. The main common features are the following:

- Use of reactor grade Pu as feed;
- High fuel burn-up;
- Possibility of including MA in the fuel for MA management.

The addition of MA to the fuel and/or blankets might constitute a proliferation resistance feature and might provide additional technical difficulty to potential proliferators. This is remarkably true in the case of radial blanket fuel elements, if present, where the produced Pu will be of reactor grade type.

Lower axial blankets in optimized spent fuel assemblies contain weapon-grade plutonium. Given the size of the foreseen axial blanket region at least 8 spent oxide FAs (or 10 spent carbide FAs) need to be diverted to obtain 1 SQ of weapon-grade plutonium. Nevertheless the possibility to misuse the reactor to irradiate fertile targets for the concealed production of undeclared Pu remains, and needs to be addressed with appropriate safeguards measures and controls.

In the case of breakout, a Sodium Fast Reactor will exhibit a large inventory of fissile material, even though not optimal for a weapons program. Nevertheless the core physics will allow a huge Pu production capability of any desired quality.

## Acknowledgments

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# INVESTING IN NEXT GENERATION SAFEGUARDS TECHNOLOGY DEVELOPMENT

**Taissa Sobolev, Marc A. Humphrey,[1] and Kevin D. Veal**

National Nuclear Security Administration (NNSA)
Department of Energy (DOE)
1000 Independence Ave, SW
Washington, D.C. 20585

***Abstract***

*The National Nuclear Security Administration of the U.S. Department of Energy (DOE/NNSA) launched the Next Generation Safeguards Initiative (NGSI) in 2008 to strengthen and sustain the international safeguards system as it evolves to meet new challenges. The NGSI's safeguards technology development subprogram focuses on transitioning promising safeguards technologies from the laboratory into the field. Since this subprogram's success depends heavily on the U.S. national laboratories' ability to develop and push the state-of-the-art with safeguards technology, NGSI has steadily increased its investment in laboratory infrastructure to better equip U.S. researchers to address international nuclear safeguards challenges. Efforts to date include the development and provision of new standards and infrastructure, and the creative use and sharing of existing materials, equipment, and facilities that help strengthen infrastructure in a cost-effective manner. This paper reviews recent NGSI efforts to identify and address the standards and infrastructure needs of the U.S. safeguards technology development community.*

**Key words:** safeguards; Next Generation Safeguards Initiative (NGSI); technology development; infrastructure; standards;

## Introduction

The National Nuclear Security Administration of the U.S. Department of Energy (DOE/NNSA) launched the Next Generation Safeguards Initiative (NGSI) in 2008 to strengthen and sustain the international safeguards system as it evolves to meet new challenges. The three objectives of the NGSI safeguards technology development subprogram are to: (1) transition advanced technologies with near-term safeguards applications from the laboratory into the field, (2) provide information management solutions to support implementation of the International Atomic Energy Agency (IAEA)'s State-Level Concept, and (3) strengthen safeguards technology development infrastructure at the National Laboratories. NGSI views the latter of these as an essential investment since this subprogram's success depends heavily on the U.S. national laboratories' ability to develop and push the state-of-the-art with safeguards technology. To date, this subprogram has funded infrastructure development projects ranging from the procurement and development of calibration standards, to the establishment of field testing facilities, and to knowledge preservation and management. In this paper we review recent and on-going NGSI efforts to identify and fulfill standards and infrastructure needs and will discuss plans to equip the U.S. national laboratories to address future safeguards challenges.

### *Standards Development*

For researchers to advance the cutting edge in technology development, they must have access to well-characterized materials and standards for experiments and calibration of equipment. NGSI

[1] Current Address: Department of Safeguards, International Atomic Energy Agency, Vienna, Austria

initiated a gap analysis survey of nuclear material in sealed sources and other standards needed by safeguards technology developers at the U.S. Department of Energy's national laboratory complex. The results of this survey have assisted in prioritizing future procurement or development of new sources, aided in evaluating the cost effectiveness of recertification or refurbishment of sources and standards, and identified the need to preserve valuable items that are in excess or slated for disposal. The survey will document the current, planned, and potential safeguards measurement and calibrations applications for which they will be used. The results of the survey will guide NGSI and other stakeholders in developing a strategic plan to ensure future access to materials of value to the safeguards technology development community.

In parallel to the gap analysis survey, NGSI is analyzing current capabilities and cost effective methods to develop new sealed sources with an emphasis on new and emerging fuel cycles. A particular focus is on actinide ceramic fuel forms and electrochemical process materials such as contaminated salts and metals. As the international community considers the development of new fuel cycles, developing technologies and equipment to safeguard these processes will be difficult without the necessary standards for model validation and instrument calibration. Priorities for these materials will be determined in coordination with the gap analysis survey and identified programmatic needs.

### *Test-bed Development*

In addition to materials and standards, tools and techniques developed in the laboratory must be tested in operational environments to demonstrate their capabilities. The NGSI safeguards technology development subprogram supports two significant infrastructure development efforts that allow researchers greater access to materials at simulated or full-scale operational facilities.

For several years, NGSI has supported the development and improvement of the Uranyl Nitrate Calibration Loop Equipment (UNCLE) Facility at the Oak Ridge National Laboratory to enable the testing and evaluation of safeguards technologies for uranium-focused fuel cycle facilities. UNCLE is designed to simulate the full-scale operating conditions of the purified aqueous uranyl nitrate solution at a natural uranium conversion plant.[i] This unique and adaptable facility provides the DOE laboratory complex with a facility to calibrate equipment and test components under controlled conditions prior to deploying to operating commercial facilities. In addition to its applicability to supporting Safeguards-by-Design (SBD), this facility has become especially important in light of continuing discussions on the starting point of safeguards. If safeguards are applied at earlier stages in the fuel cycle, UNCLE will provide a valuable opportunity for researchers to prove their technologies under a controlled environment prior to coordinating testing opportunities with operating facilities. Proving technologies in an operational setting may also help to ease operator concerns and provide more acceptability of novel safeguards technologies in their facilities.

NGSI has also supported the Nuclear Detection and Sensor Testing Center (NDSTC) at the Y-12 Nuclear Security Complex. The NDSTC was initially developed with Y-12's Plant Directed Research and Development (PDRD) program funds specifically to provide researchers with access to a wide range of materials and to support a variety of missions including nuclear nonproliferation activities.[ii] The test center has two distinct sites that provide a unique opportunity for researchers to have access to uranium-based materials, and is in a position to invite foreign nationals to participate in technology testing at one of these sites.

NGSI is working with Savannah River National Laboratory to establish a safeguards technology development test-bed at a full-scale, aqueous reprocessing facility. Built in the 1950s, the H-Canyon Facility at the Savannah River Site (SRS) is currently the only operational facility of its type in the United States. As an operating facility, the challenges it presents to safeguards technology developers closely mimic real-world challenges the IAEA faces as it tries to implement safeguards at these sensitive and complex facilities. The test-bed at H-Canyon will allow for in-line and at-line measurements and prototype instrument testing prior to demonstrations with international partners and commercial facilities. In particular, these trials will provide important information about instrument operability (e.g., in high-radiation environments), impacts on facility operations, and technology performance.

The test-bed at H-Canyon provides a wide variety of high-priority locations for testing, including the processing tanks, the transfer lines, the sampling aisle, the spent fuel receiving area, the exhaust stacks, and the high enriched uranium blend-down tanks.[iii] The implementation plan for the pilot test

period includes developing training requirements, facility access controls, technology requirements, installation paths, operating methods, and detailed costs estimates for testing advanced technologies in each location. The pilot testing phase is expected to occur in FY13- FY15.[iv] Once successfully completed, H-Canyon will be available to DOE laboratory-based technology developers. Some of the technologies selected for the pilot phase would provide for near-real-time accountancy, while others are designed to improve accuracy in accounting for plutonium in bulk handling facilities.[v,vi]

### *Material Preservation*

In additional to the physical infrastructure being considered to support safeguards technology development, NGSI has a keen interest in ensuring access to certified reference materials (CRMs) that are essential for conducting accurate safeguards measurements. NGSI has invested significantly in the preservation of ultra-high-purity U-233, a key material necessary for producing certified reference material (CRM) used in international nuclear safeguards measurements and trace-level analysis. The current supply of U-233 CRM is not expected to last more than 20 years at the projected use-rate. Materials necessary for the production of future U-233 CRM had all been slated for downblending with depleted uranium and disposition beginning in the summer of 2012. Once down-blended, this high purity material (99.5+%) will be no longer available for use in safeguards analyses. Future production of this material is highly unlikely due to the time, and cost requirements of production (which would include reconstituting dismantled facilities).

NGSI initiated a multi-year preservation effort at Oak Ridge National Laboratory (ORNL) to produce a multi-decade supply of CRM and a 50+ year supply of strategic reserve material.[vii] NGSI has been working with partners within the Department of Energy and in other U. S. government agencies to support the preservation, storage, and certification of this material. High-purity U-233 is not only essential for international nuclear safeguards, but for nuclear forensic and non-proliferation applications as well.

### *Knowledge Preservation*

NGSI is working to mitigate the pending loss of knowledge and expertise due to the retirement of some U.S. safeguards experts by documenting their specialized expertise. The goal of this knowledge-preservation effort is to capture valuable information that may reside solely with these experts so that it can be passed on to the next generation of researchers. The effort constitutes an important complement to the NGSI Human Capital Development subprogram, which aims to attract, educate, train, and retain a new generation of international safeguards professionals.

Currently, NGSI is funding a knowledge preservation effort for controlled-potential coulometry, a highly precise plutonium assay method. This analytical technique, although only used in a few, highly specialized safeguards laboratories and their staff around the world, provides extremely precise and accurate results. It is important for the characterization of CRM plutonium assay standards as well as the preparation of large size dry (LSD) spikes used by the IAEA and Japan for Isotope Dilution Mass Spectrometry (IDMS). Preserving this knowledge is key to ensuring that U.S. experts are ready and available to address future safeguards challenges in this area.

### *Inter-laboratory Resource Sharing*

In order to make safeguards-relevant materials, standards, and infrastructure at certain U.S. national laboratories available to other U.S. laboratories and researchers, NGSI has funded the development of the Safeguards Standards and Infrastructure Database[viii] hosted by Idaho National Laboratory. It is designed to facilitate the sharing of resources between laboratories to improve the cost effectiveness of safeguards technology development. For standards and other materials, this database will include detailed documentation, such as certification documentation, for each item. The database includes the capability to connect personnel across the laboratory complex to discuss opportunities to share these items on an as-needed basis. This database will provide researchers with the ability to document what they currently have and make better use of complex-wide standards, materials, and infrastructure.

### *Future Investment in Next Generation Safeguards Technology Development*

Investing in technology development infrastructure is a high priority of the NGSI Safeguards Technology Development subprogram. By providing for a strong domestic infrastructure and access to materials, U.S. safeguards technology developer will be better equipped to meet the future challenges of the international nuclear safeguards regime. NGSI will continue to work with the national laboratories for new and innovative infrastructure development or preservation projects to ensure that needs are identified and gaps addressed in parallel to developing novel safeguards technologies.

## Conclusion

NGSI is currently identifying and addressing existing gaps, while planning for future needs of next generation of safeguards technology researchers. Addressing these needs will ensure that the safeguards technology development community has the appropriate and necessary resources to develop novel technologies that address current and future challenges in international nuclear safeguards. NGSI intends to invest steadily in infrastructure at the U.S. national laboratories. Efforts to date include not only the development and provision of new standards and infrastructure, but also the creative use and sharing of existing materials, equipment, or facilities that help strengthen infrastructure in a cost-effective manner.

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# 35th Annual ESARDA Symposium on International Safeguards

## *List of participants*

| Name | Institute | Country | E-mail |
|---|---|---|---|
| Aberg-Lindell Matilde | Uppsala University Sweden | Sweden | matilda.aberg-lindell@physics.uu.se |
| Aharon Ocherashvili | Nuclear Research Center NEGEV | Israël | ocherash@bezeqint.net |
| Almási István | Hungarian Academy of Sciences Centre for Energy Research | Hungary | almasi.istvan@energia.mta.hu |
| Altmann Jürgen | Technische Universität Dortmund | Germany | altmann@e3.physik.tu-dortmund.de |
| Ancius Darius | Euratom | | darius.ancius@ec.europa.eu |
| André-Turlind Elisabeth | Swedish Radiation Safety Authority (SSM) | Sweden | elisabeth.andre.turlind@ssm.se |
| Aregbe Yetunde | EC-JRC-IRMM | Belgium | yetunde.aregbe@ec.europa.eu |
| Avenhaus Rudolf | Universität der Bundeswehr München | Germany | rudolf.avenhaus@unibw.de |
| Axell Kare | SSM | Sweden | kare.axell@ssm.se |
| Bagi János | Centre for Energy Research, Hungarian Academy of Sciences | Hungary | janos.bagi@energia.mta.hu |
| Barefield II James | Los Alamos National Laboratory | United States | jbarefield@lanl.gov |
| Bartels Erik | Ministerie van Infrastructuur en Milieu | Netherlands | erik.bartels@ilent.nl |
| Bayle Marcel | Comité Technique Euratom | France | marcel.bayle@cte.gouv.fr |
| Beaman Michael | Health and Safety Executive - Office of Nuclear Regulation | United Kingdom | mike.beaman@hse.gsi.gov.uk |
| Becker Bjorn | EC-JRC-IRMM | Belgium | bjorn.becker@ec.europa.eu |
| Benz Jacob | Pacific Northwest National Laboratory | United States | jacob.benz@pnnl.gov |
| Bernstein Adam | Lawrence Livermore National Laboratory | United States | bernstein3@llnl.gov |
| Black Claudie | European Commission, Joint Research Centre, ITU | Germany | Claudie-Kate.Black@ec.europa.eu |
| Bolind Alan | Japan Atomic Energy Agency | Japan | alan.bolind@jaea.go.jp |
| Bon Nguyen Romuald | IRSN | France | romuald.bonnguyen@irsn.fr |

| | | | |
|---|---|---|---|
| Borella Alessandro | SCK•CEN | Belgium | aborella@sckcen.be |
| Bosko Andrey | Canberra Industries Inc. | United States | abosko@canberra.com |
| Bourva Ludovic | Canberra | United Kingdom | ludovic.bourva@canberra.com |
| Brassinnes Matilda | Uppsala University | Sweden | matilda.aberg-lindell@physics.uu.se |
| Brown Simon | Symetrica Security Ltd | United Kingdom | s.brown@symetrica.com |
| Buják Renáta | EC- JRC-IRMM | Belgium | renata.bujak@ec.europa.eu |
| Burjan Atilla | AWE | United Kingdom | atilla.burjan@awe.co.uk |
| Burrows Brian | BNC Limited | United Kingdom | BRIAN.BURROWS10@NTLWORLD.COM |
| Button Peter | Canadian Nuclear Safety Commission (CNSC) | Canada | peter.button@cnsc-ccsn.gc.ca |
| Cartemo Petty | Chalmers University of Technology | Sweden | petty@nephy.chalmers.se |
| Chandra Rico | Arktis Radiation Detectors Ltd | Switzerland | rico.chandra@arktis-detectors.com |
| Chaptal-Gradoz Nathalie | IRSN | France | nathalie.chaptal-gradoz@irsn.fr |
| Chard Patrick | CANBERRA UK LIMITED | United Kingdom | patrick.chard@canberra.com |
| Charlton William | Texas A+ M University | United States | wcharlton@tamu.cdu |
| Chater Simon | Sellafield Ltd | United Kingdom | simon.p.chater@sellafieldsites.com |
| Chernikova Dina | Chalmers University of Technology | Sweden | dina@nephy.chalmers.se |
| Chorna Nadiia | State Nuclear Regulatory Inspectorate of Ukraine | Ukraine | chorna@hq.snrc.gov.ua |
| Chumak Dmytro | Odessa I.I.Mechnikov National University | Austria | chumak.dmytro@gmail.com |
| Ciccarello Stefano | Euratom | | stefano.ciccarello@ec.europa.eu |
| Clarke Nicholas | Sellafield Ltd | United Kingdom | nicholas.i.clarke@sellafieldsites.com |
| Cojazzi Giacomo G.M. | EC, JRC, Institute for Transuranium Elements | Italy | giacomo.cojazzi@jrc.ec.europa.eu |
| Colin Alexandra | CEA | France | alexandra.colin@cea.fr |
| Collet Aurélien | EU Satellite Center | Spain | aurelien.collet@satcen.europa.eu |
| Communaux Michel | Electricté de France | France | michel.communaux@edf.fr |
| Corcho José | SPIEZ LABORATORY | Switzerland | jose.corcho@babs.admin.ch |
| Croft Stephen | ORNL | United States | CroftS@ornl.gov |

| | | | |
|---|---|---|---|
| Crozes- Madec Diane | IRSN | France | diane.crozes-madec@irsn.fr |
| Dahlberg Joakim | SSM | Sweden | Joakim.Dahlberg@ssm.se |
| Damge Eric | AREVA | France | eric.damge@areva.com |
| Dams Chris | FANC | Belgium | chris.dams@fanc.fgov.be |
| Daniel Guillaume | IRSN | France | guillaume.daniel@irsn.fr |
| Darko Korosec | Intrernational Atomic Energy Agency | Austria | d.korosec@iaea.org |
| De Baerdemaeker Luc | CANBERRA | Belgium | anja.vandenborre@canberra.com |
| Denot Thierry | Electricité de France | France | thierry.denot@edf.fr |
| Dreicer Mona | Lawrence Livermore National Laboratory | United States | dreicer1@llnl.gov |
| Dresselaers Rony | FANC | Belgium | rony.dresselaers@fanc.fgov.be |
| Dufva Martina | Swedish Radiation Safety Authority | Sweden | martina.dufva@ssm.se |
| Dürr Martin | Forschungszentrum Juelich | Germany | ma.duerr@fz-juelich.de |
| Edmonds Nick | Sellafield Ltd | United Kingdom | nick.l.edmods@sellafieldsites.com |
| Eigenbrodt Julia | Los Alamos National Laboratory | United States | jmeigenbrodt@gmail.com |
| Englert Matthias | IANUS, TU Darmstadt | Germany | englert@ianus.tu-darmstadt.de |
| Erdmann Nicole | European Commission JRC | Germany | nicole.erdmann@ec.europa.eu |
| Fallot Muriel | SUBATECH laboratory - Univ. of Nantes | France | fallot@subatech.in2p3.fr |
| Fanchini Erica | INFN Sezione di Genova | Italy | fanchini@ge.infn.it |
| Fanielle Sylvain | Cabinet of the Belgian Minister of Foreign Affairs | | s.fanielle@gmail.com |
| Fazly Oksana | State Nuclear Regulatory Inspectorate of Ukraine | Ukraine | fazly@hq.snrc.gov.ua |
| Fernandez Moreno Sonia | ABACC | Brazil | winarni@abacc.org.br |
| Francis Stephen | UK National Nuclear Laboratory | United Kingdom | stephen.m.francis@nnl.co.uk |
| Funk Pierre | IRSN | France | pierre.funk@irsn.fr |
| Gao Henry | Canadian Nuclear Safety Commission | Canada | henry.gao@cnsc-ccsn.gc.ca |
| Georg Uwe | Swiss Federal Office of Energy | Switzerland | uwe.georg@bfe.admin.ch |
| Giorgiantoni Giorgio | ENEA | Italy | giorgio.giorgiantoni@enea.it |

| | | | |
|---|---|---|---|
| Goddard Braden | NSSPI, Texas A&M University | United States | goddard.braden@gmail.com |
| Göttsche Malte | Centre for Science and Peace Research, Univ Hamburg | Germany | malte.goettsche@physik.uni-hamburg.de |
| Granier Guy | CEA Marcoule | France | guy.granier@cea.fr |
| Grape Sophie | Uppsala University | Sweden | sophie.grape@gmail.com |
| Harada Hideo | Japan Atomic Energy Agency (JAEA) | Japan | harada.hideo@jaea.go.jp |
| Hardeman Frank | SCK•CEN | Belgium | frank.hardeman@sckcen.be |
| Hartigan Kelsey | Nuclear Threat Initiative | United States | hartigan@nti.org |
| Hayakawa Takehito | Japan Atomic Energy Agency | Japan | hayakawa.takehito@jaea.go.jp |
| Holzleitner Ludwig | European Commission, ITU | Germany | Ludwig.Holzleitner@ec.europa.eu |
| Homer Alan | Sellafield Ltd | United Kingdom | alan.homer@sellafieldsites.com |
| Horak Karl | Sandia National Laboratories | United States | kehorak@sandia.gov |
| Hustveit Styrkaar | Norwegian Radiation Protection Authority | Norway | styrkaar.hustveit@nrpa.no |
| Huszti József | Centre for Energy Research | Hungary | huszti.jozsef@energia.mta.hu |
| Jacobsson Svärd Staffan | Uppsala University | Sweden | staffan.jacobsson_svard@physics.uu.se |
| Jalouneix Jean | IRSN | France | jean.jalouneix@irsn.fr |
| James Garner | Oak Ridge National Laboratory | United States | garnerjr@ornl.gov |
| Janssens Willem | JRC | Italy | willem.janssens@ec.europa.eu |
| Jansson Peter | Uppsala University | Sweden | peter.jansson@physics.uu.se |
| Jarmalaviciute Jurgita | State Nuclear Power Safety Inspectorate | Lithuania | jurgita.jarmalaviciute@vatesi.lt |
| Jonter Thomas | Stockholm University | Sweden | thomas.jonter@ekohist.su.se |
| Jussofie Astrid | GNS Gesellschaft für Nuklear Service mbH | Germany | astrid.jussofie@gns.de |
| Kahnmeyer Wolfgang | Euratom | | wolfgang.kahnmeyer@ec.europa.eu |
| Keir Allen | AWE | United Kingdom | keir.allen@awe.co.uk |
| Keir David | VERTIC | | |
| Kharchaf Amina | University Ibn Tofail | Morocco | akharchaf@yahoo.fr |
| Kim Ho-Dong | Korea Atomic Energy | Korea (South) | khd@kaeri.re.kr |

| | | | |
|---|---|---|---|
| Kivel Niko | Paul Scherrer Institute | Switzerland | niko.kivel@psi.ch |
| Knott Alexander | Research Centre Juelich GmbH | Germany | a.knott@fz-juelich.de |
| Köble Theo | Fraunhofer-INT | Germany | theo.koeble@int.fraunhofer.de |
| Koblet Elisabeth | Swedish Radiation Safety Authority | Sweden | elisabeth.koblet@ssm.se |
| Korbmacher Tim | URENCO Ltd. | Germany | Tim.Korbmacher@URENCO.com |
| Kory Budlong Sylvester | Los Alamos National Laboratory | United States | ksylvester@lanl.gov |
| Koutsoyannopoulos Christos | Euratom | | christos.koutsoyannopoulos@ec.europa.eu |
| Kovács-Széles Éva | Centre for Energy Research, Hungarian Academy of Sc iences | Hungary | kovacs-szeles.eva@energia.mta.hu |
| Kraiem Monia | U.S. Department of Energy (DOE) | United States | monia.kraiem@ch.doe.gov |
| Kruk Krzysztof | National Atomic Energy Agency | Poland | krzysztof.kruk@paa.gov.pl |
| Kütt Moritz | IANUS, Technische Universität Darmstadt | Germany | kuett@ianus.tu-darmstadt.de |
| Laas Roman | Tomsk Polytechnic University | Russian Federation | roman.laas@gmail.com |
| Lafitte Marc | European Union Satellite Centre | Spain | marc.lafitte@satcen.europa.eu |
| Lahogue Yves | Euratom | | yves.lahogue@ec.europa.eu |
| Lahti Mari | Posiva Oy | Finland | mari.lahti@posiva.fi |
| Lee Andrew George | Sellafield Ltd | United Kingdom | andrew.g.lee@sellafieldsites.com |
| Leffer Teri | U.S. Department of Energy, National Nuclear Security Administration | United States | westerfeldt@msn.com |
| Liljenfelt Henrik | Svensk Kärnbränslehanterng AB | Sweden | |
| Listner Clemens | Forschungszentrum Jülich | Germany | c.listner@fz-juelich.de |
| Loreaux philippe | CEA | France | philippe.loreaux@cea.fr |
| Lythgoe Clive | Sellafield Limited | United Kingdom | clive.lythgoe@sellafieldsites.com |
| MacArthur Duncan | Los Alamos National Laboratory | United States | dmacarthur@lanl.gov |
| Macsik Zsuzsanna | IAEA | Austria | Z.Macsik@iaea.org |
| Marcuzzo Do Canto Odilon | ABACC | Brazil | winarni@abacc.org.br |
| McDaniel Michael | Sandia National Laboratories | United States | mmcdani@sandia.gov |
| Medici Fausto | Federal office of Energy | Switzerland | fausto.medici@bfe.admin.ch |

| | | | |
|---|---|---|---|
| Mermaz Frederic | IRSN | France | frederic.mermaz@irsn.fr |
| Meylemans Paul | Euratom | | paul.meylemans@ec.europa.eu |
| Mladineo Stephen | Pacific Northwest National Laboratory | United States | steve.mladineo@pnnl.gov |
| Mochiji Toshiro | Japan Atomic Energy Agency | Japan | mochiji.toshiro@jaea.go.jp |
| Morin Maxime | IRSN | France | maxime.morin@irsn.fr |
| Moring Mikael | STUK | Finland | Mikael.Moring@stuk.fi |
| Morris Thomas | Sellafield Ltd | United Kingdom | thomas.a.morris@sellafieldsites.com |
| Muehleisen Artur | EC JRC ITU | Germany | artur.muehleisen@ec.europa.eu |
| Murer David | Arktis Radiation Detectors Ltd | Switzerland | murer@arktis-detectors.com |
| Murtezi Mentor | Euratom | | mentor.murtezi@ec.europa.eu |
| Nazarenko Anton | Baltic Scientific Instruments | Latvia | sales@bsi.lv |
| Neuneck Götz | IFSH | | |
| Niemeyer Irmgard | Forschungszentrum Jülich GmbH | Germany | i.niemeyer@fz-juelich.de |
| Nussbaum Sven | Center for Remote Sensing of Land Surfaces, Univ. of Bonn | Germany | snussbaum@nussbaumnet.de |
| Ocherashvili Aharon | NRCN | Israel | |
| Oddou Julie | Comité Technique Euratom | France | julie.oddou@cea.fr |
| Parcey Dennis | Channel Systems Inc. | Canada | dennis.parcey@channelsystems.ca |
| Peerani Paolo | European Commission - Joint Research Centre | Italy | paolo.peerani@jrc.ec.europa.eu |
| Peixoto Orpet | ABACC | Brazil | winarni@abacc.org.br |
| Persson Karin | Uppsala University | Sweden | karin.persson@uadm.uu.se |
| Pickett Chris | Oak Ridge National Laboratory | United States | cvpickett@comcast.net |
| Pilat Joseph | Los Alamos National Laboratory | United States | sylvia@lanl.gov |
| Pointurier Fabien | CEA | France | fabien.pointurier@cea.fr |
| Quaglia Bruno | Comité Technique Euratom | France | bruno.quaglia@cte.gouv.fr |
| Reistad Ole | | | |
| Rezniczek Arnold | UBA Unternehmensberatung GmbH | Germany | rezniczek@uba-gmbh.de |

| | | | |
|---|---|---|---|
| Richard Michel | Individual member | | micheljrichard@orange.fr |
| Richter Stephan | IRMM/JRC | Belgium | stephan.richter@ec.europa.eu |
| Richter Bernd | FORSCHUNSGSZENTRUM JÜLICH | Germany | bum.juelich@gmx.net |
| Rosenstock Wolfgang | Fraunhofer - INT | Germany | wolfgang.rosenstock@int.fraunhofer.de |
| Rossa Riccardo | SCK•CEN | Belgium | rrossa@sckcen.be |
| Satkowiak Lawrence | Oak Ridge National Laboratory | United States | satkowiaklj@ornl.gov |
| Sbaffoni M. Monica | IAEA | Austria | m.sbaffoni@iaea.org |
| Schaper Annette | PRIF | | |
| Schillebeeckx Peter | EC - JRC - IRMM | Belgium | peter.schillebeeckx@ec.europa.eu |
| Schoop Konrad | Euratom | | konrad.schoop@ec.europa.eu |
| Schwalbach Peter | Euratom | | peter.schwalbach@ec.europa.eu |
| Sednev Dmitry | Tomsk polytechnic university | Russian Federation | sednev@tpu.ru |
| Sekse Tonje | Norwegian Radiation Protection Authority | Norway | tonje.sekse@nrpa.no |
| Sevini Filippo | European Commission, Joint Research Centre | Italy | filippo.sevini@jrc.ec.europa.eu |
| Seya Michio | Japan Atomic Energy Agency | Japan | seya.michio@jaea.go.jp |
| Shizuma Toshiyuki | Japan Atomic Energy Agency | Japan | shizuma.toshiyuki@jaea.go.jp |
| Simpson Alan | Pajarito Scientific Corporation | United Kingdom | asimpson@pscnda.com |
| Sitdikova Alexandra | Tomsk Polytechnic University | Russian Federation | alexandra.sitdikova@gmail.com |
| Sjöland Anders | SKB AB | Sweden | anders.sjoland@skb.se |
| Smartt Heidi | Sandia National Labs | United States | hasmart@sandia.gov |
| Smith L. Eric | International Atomic Energy Agency | Austria | e.smith@iaea.org |
| Sobolev Taissa | U.S. Department of Energy | United States | taissa.sobolev@nnsa.doe.gov |
| Song Kyuseok | Korea Atomic Energy Research Institute | Korea, Republic Of | sks@kaeri.re.kr |
| Sorenson Ken D. | Sandia National Laboratories | United States | ksoren@sandia.gov |
| Stein Gotthard | Consultant | Germany | g.stein@fz-juelich.de |
| Sturm Monika | Division of Analytical Chemistry – VIRIS | Austria | sturm.monika@gmail.com |

| | | | |
|---|---|---|---|
| Sundén Erika | Swedish Radiation Safety Authority | Sweden | erika.sunden@ssm.se |
| Svedkauskaite Jolanta | EC, EG ENER | Luxembourg | jolanta.svedkauskaite@ec.europa.eu |
| Synetos Sotiris | Euratom | | sotiris.synetos@ec.europa.eu |
| Tagziria Hamid | European Commission – JRC - ITU- | Italy | hamid.tagziria@jrc.ec.europa.eu |
| Tobin Stephen | LANL | United States | tobin@lanl.gov |
| Toivonen Harri | STUK | Finland | harri.toivonen@stuk.fi |
| Toma Magdalena | EC - JRC - IRMM | Germany | Magdalena.Toma@ec.europa.eu |
| Trautwein Wolfgang | Federal Ministry of Economics and Technologiy | Germany | wolfgang.trautwein@bmwi.bund.de |
| Treppmann Christoph | Enritec | | Christoph.Treppmann@ext.enritec.com |
| Truyens Jan | EC – JRC - IRMM | Belgium | jan.truyens@ec.europa.eu |
| Tsuchiya Harufumi | JAEA | Japan | tsuchiya.harufumi@jaea.go.jp |
| Tsvetkov Igor | IAEA | Austria | i.tsvetkov@iaea.org |
| Turza Peter | Public Limited Company for Radioactive Waste | Hungary | turza.peter@rhk.hu |
| Tushingham James | National Nuclear Laboratory | United Kingdom | james.tushingham@nnl.co.uk |
| Twomey Tim | ORTEC | Ireland | Tim.Twomey@ametek.com |
| Vacheret Antonin | University of Oxford | United Kingdom | Antoninv@gmail.com |
| van der Meer Klaas | SCK•CEN | Belgium | kvdmeer@sckcen.be |
| Van Severen Uwe Joerg | ORTEC | Germany | vanseveren@ametek.de |
| Vincze Arpad | Hungarian Atomic Energy Authority | Hungary | vincze@haea.gov.hu |
| Voute Frederik | Erasmus University Rotterdam/Clingendael Institute | Netherlands | fvoute@gmail.com |
| Weber Anne-Laure | IRSN | France | anne-laure.weber@irsn.fr |
| Whichello Julian | International Atomic Energy Agency | Austria | j.whichello@iaea.org |
| Whitaker Michael | Oak Ridge National Laboratory | United States | Whitakerjm@ornl.gov |
| White Helen | AWE | United Kingdom | helen.white@awe.co.uk |
| Wolfart Erik | EC-JRC | Italy | erik.wolfart@jrc.ec.europa.eu |
| Woowat Clare | Sellafield Ltd | United Kingdom | clare.woowat@sellafieldsites.com |

| | | | |
|---|---|---|---|
| Yamaguchi Tomoki | Japan Atomic Energy Agency | Japan | yamaguchi.tomoki@jaea.go.jp |
| Zsigrai Jozsef | Institute for Transuranium Elements | Germany | jozsef.zsigrai@ec.europa.eu |
| Zykov Sergey | IAEA | | |

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