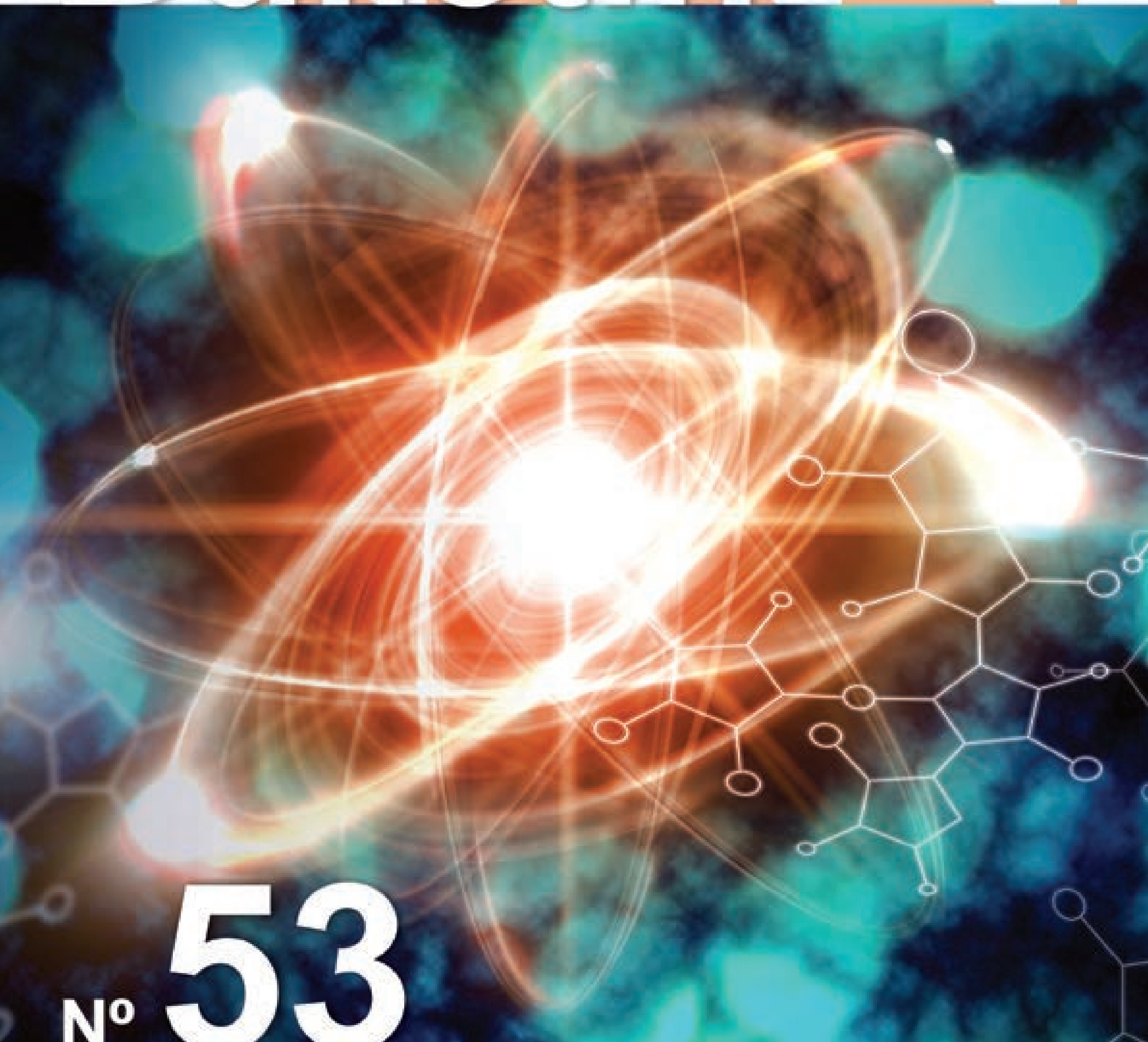
ISSN 1977-5296

ESARDA

European Safeguards Research and Development Association

Bulletin



Nº 53

December 2015

ISSN 1977-5296

Number 53
December 2015

Editor
Hamid Tagziria
European Commission, Joint Research Centre,
ITU - Nuclear Security Unit
T.P. 800, I-21027 Ispra (VA), Italy
Tel. +39 0332-786324
esarda-bulletin@jrc.ec.europa.eu
hamid.tagziria@jrc.ec.europa.eu

ESARDA is an association formed to advance and harmonize research and development for safeguards. The Parties to the association are:

Areva, France; ATI, Austria; CEA, France; CNCAN, Romania; EDF, France; ENEA, Italy; European Commission; FZJ, Germany; HAEA, Hungary; MTA EK, Hungary; IRSN, France; MINETUR, Spain; NNL, UK; NRI, Czech Republic; NRPA, Norway; SCK/CEN, Belgium; Sellafield Ltd, UK; SFOE, Switzerland; SSM, Sweden; Springfields Fuels Ltd, UK; ST, Finland; University of Hamburg, Germany; University of Liege, Belgium; University of Uppsala, Sweden; AEA, UK; URENCO, Germany; VATESI, Lithuania; WKK, Germany; PAA, Poland; ORNL, USA

Scientific and technical papers submitted for publication in the peer reviewed section are reviewed by independent authors including members of the Editorial Committee.

Manuscripts are to be sent to the Editor (esarda-bulletin@jrc.ec.europa.eu) following the 'instructions for authors' available on https://esarda.jrc.ec.europa.eu/ where the bulletins can also be viewed and downloaded.

Photos or diagrams should be of high quality.

Accepted manuscripts are published free of charge.

N.B. Articles and other material in the ESARDA Bulletin do not necessarily present the views or policies of neither ESARDA nor the European Commission.

ESARDA Bulletin is published jointly by ESARDA and the Joint Research Centre of the European Commission and distributed free of charge to over 1100 registered members, libraries and institutions Worldwide.

The publication is authorized by ESARDA.

Cover designed by Laura Spirito (JRC Ispra in Italy), using illustration of Ezume Images (Fotolia)

Printed by
IMPRIMERIE CENTRALE, Luxembourg

ESARDA
European Safeguards Research and Development Association

Bulletin

Table of Content Issue n° 53

Editorial

Hamid Tagziria 1

Peer Reviewed Articles

Comparison of fresh fuel experimental measurements to MCNPX results using the differential die-away instrument for nuclear safeguards applications 2
Alison V. Goodsell, Vladimir Henzl, Martyn T. Swinhoe, Carlos Rael, David Desimone, William S. Charlton

A Qualitative Analysis of the Neutron Population in Fresh and Spent Fuel Assemblies during Simulated Interrogation using the Differential Die-Away Technique 13
Niklas Lundkvist, Alison V. Goodsell, Sophie Grape, John S. Hendricks, Vladimir Henzl, Martyn T. Swinhoe and Stephen J. Tobin

Improving the prediction model for Cherenkov light generation by irradiated nuclear fuel assemblies in wet storage for enhanced partial-defect verification capability 22
Erik Branger, Sophie Grape, Peter Jansson, Staffan Jacobsson Svärd

Experimental Assessment of a LiF:ZnS(Ag) Prototype Neutron Coincidence Counter for Safeguards 30
H. Tagziria, M. Foster, M. Schear, D. Ramsden, G. Dermody, B. Pedersen, and P. Schwalbach

Detection of fission signatures induced by a low-energy neutron source 47
A. Ocherashvili, V. Mayorov, A. Beck, G. Heger, E. Roesgen, J.-M. Crochemore, M. Mosconi, B. Pedersen

Emerging Applications of Bottom-Up Uncertainty Quantification in Nondestructive Assay 54
Burr, T., Croft, S., Dale, D., Favalli, A., Weaver, B., Williams, B.

Mobile 3D Laser Scanning for Nuclear Safeguards 62
E. Wolfart, S. Ceriani, D. Puig, C. Sanchez, P. Taddei, V. Sequeira

Monitoring Uranium Mining and Milling using Commercial Observation Satellites 73
Lalitha Sundaresan, Chandrashekar Srinivasan and Bhupendra Jasani

Systems Approach to Arms Control Verification 83
Keir Allen, Mona Dreicer, Cliff Chen, Irmgard Niemeyer, Clemens Listner, Gotthard Stein

Safeguards-relevant information collection from small holders – experiences and challenges 92
Erika Sundén, Martina Dufva, Joakim Dahlberg

Methodological Aspects of the IAEA State Level Concept and Acquisition Path Analysis: A State's Nuclear Fuel Cycle, Related Capabilities, and the Quantification of Acquisition Paths 100
Lance K. Kim, Guido Renda, Giacomo G. M. Cojazzi

Other Articles

Advancing Disarmament Verification Tools: A Task for Europe? 111
Malte Göttsche, Moritz Kütt, Götz Neuneck, Irmgard Niemeyer

Editorial

Hamid Tagziria

Dear Readers,

The International Partnership for nuclear disarmament verification (IPNDV) had its 2nd plenary session in Oslo from 16 to 18 November, opened by the Foreign Minister of Norway Borge Brende.

Co-organised by the USA and Norway, it was attended by 23 national delegations (in addition to delegates from VERTIC, CTBTO and OPCW):

- The Nuclear Non-Proliferation Treaty (NPT) Nuclear-Weapon States (NWS), or P5: France, UK, USA, China, Russia,
- 9 EU Member States,
- 11 non-EU non Nuclear-Weapon States (NNWS)
- The European External Action Service (EEAS) supported by DG-ENER and DG-JRC

IPNDV broadly aims to:

- Build international capacity amongst NWS and non-NWS
- Improve and broaden the understanding of the challenges faced within nuclear disarmament verification and monitoring
- Provide international leadership by facilitating technical progress to meet these challenges

Three working groups (Monitoring and verification objectives, On-site inspections and Technical challenges and solutions) which had previously been formed have now finalised and agreed their terms of reference.

Seeking to lay solid foundations for further reductions in nuclear weapons and advance NPT nuclear disarmament goals it was acknowledged that the monitoring and verification issues across the full nuclear lifecycle (from production of fissile material to disposal after dismantlement), i.e. including irreversibility should be assessed. However, it was argued that in order to be effective, ensure consensus and be able to show tangible results at the end of an 18 months period it was decided that the initial focus of IPNDV will be on the monitoring and verification of nuclear warhead dismantlement and the nuclear material which result from the dismantled Nuclear Warheads.

The plenary and working group meetings provided interesting and valuable information such as lessons learned from a number of initiatives (UK-USA, UK-Norway, USA-Russia), comparing different inspection regimes, understanding other verification and monitoring R&D and understanding safety and security requirements and boundary conditions (information barriers etc.) for the verification of nuclear disarmament.

Nuclear Disarmament and Arms control are fundamentally dependant on verification which in turn heavily depends on a variety of techniques and methods that are well known, used and studied by the ESARDA communities and working groups generally (NDA, C&S, VTM, NA/NT..). It is therefore natural that initiatives such as the IPNDV in Disarmament verification should be able to find in ESARDA a good partner, forum and platform to further progress, develop synergies and collaborations and help towards a more comprehensive European engagement in particular as well as word wide. This was recognised during the panel discussion of the 36th ESARDA symposium in Bruges (2013) on the subject of *Disarmament Verification - A dialogue on Technical and Transparency Issues* as reported in issue 50 of the ESARDA bulletin [3]. Furthermore an INMM - ESARDA collaboration would be most beneficial to the IPNDV and to any other initiative of its kind. On the European level, the EU council established (decision 2010/430/CFSP) a non-proliferation consortium (think tank) with the aim "*to encourage discussion of measures to combat the proliferation of weapons of mass destruction and their delivery systems within civil society, particularly among experts, researchers and academics*".

Hamid Tagziria
Editor and Editorial Committee Chairman

https://esarda.jrc.ec.europa.eu
Esardabulletin@jrc.ec.europa.eu
hamid.tagziria@jrc.ec.europa.eu

References:

[1] www.ipndv.org and http://www.nti.org (IPNDV)
[2] http://ukni.info/ (NWS-NNWS, UK-Norway Initiative)
[3] http://esarda.jrc.ec.europa.eu (ESARDA Bulletin, no. 50, December 2013)
[4] http://www.nonproliferation.eu/

Comparison of fresh fuel experimental measurements to MCNPX results using the differential die-away instrument for nuclear safeguards applications

Alison V. Goodsell[1,2],* Vladimir Henzl[1], Martyn T. Swinhoe[1], Carlos Rael[1], David Desimone[1], William S. Charlton[2]

[1] Los Alamos National Laboratory
P.O. Box 1663, Los Alamos, NM 87545, USA
[2] Nuclear Security Science & Policy Institute
Texas A&M University, College Station, TX 77840, USA
E-mail: alison.goodsell@pnnl.gov, henzl@lanl.gov

Abstract:

A Differential Die-Away instrument, based on the Differential Die-Away (DDA) non-destructive assay technique, is currently being investigated at Los Alamos National Laboratory (LANL) to better understand its capabilities and deployment challenges. The DDA instrument is based on an active neutron interrogation technique which uses an external deuterium-tritium (DT) neutron generator to induce fission in a fuel assembly. The time of arrival (list-mode data) of the prompt fission neutrons are detected by nine ^{3}He detectors positioned around the fuel assembly. The characteristics of an assayed fuel assembly, such as the fissile content and presence of neutron absorbers, impact the time required for the DDA signal to die away. Previously performed spent fuel Monte Carlo N-Particle eXtended transport code (MCNPX) simulations have shown that the dynamic evolution of the DDA signal can reveal various characteristics of a spent fuel assembly. Based on these simulations, the principal DDA instrument capabilities include measurement of multiplication, total effective fissile mass, total plutonium content, estimation of basic fuel assembly parameters such as initial enrichment and burnup, and identification of certain partial defects. In this work, we aim to support and complement the previous simulation-only based, yet promising, results with experimental measurements using fresh pressurized water reactor (PWR) nuclear fuel in a water tank. While the isotopic simplicity of fresh fuel is not comparable to isotopic complexity of the spent fuel, we expect the results of this study to confirm general trends reflecting the overall physics of the DDA instrument as previously identified. Such assumption is based on the fact that the DDA signal reflects the effective fissile and neutron absorber content, not a particular isotopic composition. Moreover, we compare the primary experimental observables, such as magnitude and die-away time of the DDA signal, with results of new dedicated MCNPX simulations. This comparison tests the reliability of the MCNPX code for simulation of nuclear fuel assay by the DDA method and is thus critical in the development and possible deployment of the DDA instrument. We present results of our efforts to identify possible discrepancies and to quantify sources of uncertainty in the experiment and the simulation.

Keywords: differential die-away, nuclear fuel, non-destructive assay, safeguards, active interrogation

1. Introduction

The differential die-away (DDA) instrument is an active, non-destructive assay neutron interrogation technique for nuclear safeguards applications currently being investigated at Los Alamos National Laboratory (LANL). The instrument uses short pulses from an external deuterium-tritium (DT) neutron generator to induce fission in the fissile material of a fuel assembly. The DDA signal magnitude and dynamic evolution characterized by the die-away time depends on the amount of fissile material and neutron absorbers in the fuel. The time-dependent signal is recorded using a list-mode data acquisition system by multiple helium-3 (^{3}He) detectors positioned around the fuel assembly.

Spent fuel simulations using Monte Carlo N-Particle eXtended (MCNPX) transport code [1] performed under the US National Nuclear Security Administration's Defense Nuclear Nonproliferation Next Generation Safeguards Initiative Spent Fuel project (NGSI-SF) [2] showed the capability of the DDA instrument to characterize properties of a wide range of spent fuel assemblies (SFAs) through analysis of the magnitude and die-away time of the DDA signal [3]. Based on results of MCNPX simulations, the predicted DDA instrument capabilities include determination of SFAs multiplication, burnup and initial enrichment, as well as the total fissile content, effective fissile mass, and identification of certain types of partial defects [4].

While the DDA technique has been used to assay drums of nuclear waste for storage [5] [6] [7] [8] the recently proposed application that is investigated within this paper faces many different challenges. Compared to the traditional use, we are investigating the application of the DDA technique in the time domains nearly directly after the interrogating neutron pulse, assaying a highly radioactive and highly multiplicative item with the moderator (i.e. water) distributed in the fixed item matrix.

The primary motivation for the experimental campaign with fresh fuel is to validate the general conclusions based on the previously performed simulations of spent fuel assay [4]. While the isotopic composition of the spent fuel is far

* A.V. Goodsell is now affiliated with Pacific Northwest National Laboratory W.S. Charlton is now affiliated with the National Strategic Research Institute at University of Nebraska.

more complex than that of the fresh fuel, the dynamic properties of the DDA signal are not sensitive to a particular isotopic composition, but rather to the effective content of fissile material and neutron absorbers. Therefore high-risk operations with highly radioactive and hard to access spent fuel are considered not to be necessary in order to verify relative properties of the DDA signal with respect to many assembly parameters (e.g. enrichment, presence of neutron absorbers, specific interrogation scenario, etc.). Moreover, new dedicated MCNPX simulations that are directly comparable to experimental measurements can verify the reliability of the code in terms of simulating the complex process of active assay of a highly multiplying item in form of nuclear fuel assembly. This comparison, if sufficiently equivalent, may lend further credibility to the previous simulations where spent fuel was assayed. Additional objectives achievable with the fresh fuel measurement campaign include verification of detector and data acquisition system performance in a high rate environment and providing relevant information and best practices for the development and eventual deployment of a DDA instrument for spent fuel characterization and verification.

2. Fresh Fuel Experimental Setup

2.1 DDA Components

The LANL laboratory-scale prototype DDA instrument consists of nine ^{3}He detectors inside three stainless steel detector pods, and an external DT neutron generator inside of a waterproof cylinder (Fig. 1). These components are all submerged in a water tank. A template made of high-density polyethylene (HDPE) is positioned on the bottom of the water tank to align the experimental components. A second HDPE template is used to position the detectors, which are inserted into HDPE cylinders wrapped with cadmium (Cd), inside the stainless steel pods. Neutron detection data from individual detectors are recorded using a list-mode data acquisition system such that the time of arrival of each pulse is recorded.

2.2 Fresh Fuel and Assembly Specifications

Fresh fuel rods containing uranium dioxide (UO_2) were used for the active assay with the DDA instrument. A 15x15 PWR-like fuel lattice with 204 fuel pin positions and 21 guide tubes was used. The fresh fuel at LANL consists of low-enriched uranium (LEU) and depleted uranium (DU) fuel pins with enrichments of approximately 3.12% to 3.25% and 0.20% to 0.24% ^{235}U, respectively. Different combinations of LEU and DU rods were used to create fuel assembly with a particular effective (i.e. average) enrichment such as those used in this study. The fuel assembly and fuel pin specifications are provided in Table I.

PWR Assembly		
Lattice geometry	15 x 15	
Assembly width	21.5 cm	
Fuel pin pitch	1.4 cm	
Number of fuel pin slots	204	
Number of guide tube slots	21	
Fuel Pin Information		
Fuel type	UO_2	
Cladding type	Zircaloy-2	
Average rod enrichment		
LEU	3.19% ^{235}U	
DU	0.22% ^{235}U	
Fuel pellet density	10.48 g/cm^3	
Fuel pellet radius	0.4525 cm	
Cladding thickness	0.0875 cm	
Outer pin radius	0.54 cm	
Total fuel rod length	130 cm	
Active fuel length		
LEU rod	102 cm	
DU rod	120 cm	
Inert fuel regions	LEU	DU
Top	17 cm	6 cm
Bottom	12 cm	5 cm

Table I: LANL PWR 15x15 fresh fuel and assembly specifications.


Figure 1: The DDA instrument setup with the experimental components slotted into the base template (left) and the setup submerged in the water tank (right).

2.3 DT Neutron Generator

The Thermo Scientific P 385 deuterium-tritium (DT) neutron generator used in this work produces an approximate maximum yield of $5{\cdot}10^8$ n/s of 14.1 MeV neutrons from DT fusion. During the experimental campaign, the neutron generator has been operated within standard manufacturer recommended parameters at 125 kV or 90 kV high voltage, 70 µA beam current, 5% or 10% duty cycle, and 2500 Hz pulse frequency. The neutron generator output is monitored using a 12” active length ^{3}He detector inserted into a HDPE block and wrapped in a Cd liner. This flux monitor is currently positioned in a fixed location adjacent to the NG outside of the water tank. The flux monitor data are compared between experimental runs to verify the consistency of the neutron generator output over time.

2.4 Detectors and Electronics

Nine 12” active length, 4 atmosphere pressure ^{3}He tubes with a fill gas of argon/methane ($Ar\text{-}CH_4$) or carbon dioxide (CO_2) operated at 1800 V were used in the fresh fuel DDA experiments. The ^{3}He detectors were paired with PDT-10A pre-amplifiers with an AMPTEK A-111 chip for fast pulse processing [9]. These pre-amplifiers are specifically designed to operate in high count rate environments.

2.5 List-Mode Data Acquisition System and Analysis

List-mode data are acquired using a data acquisition system assembled at LANL from all commercially available parts from National Instruments, including a Portable Monitor Accessory (NI PMA-1115) with PXI Express Chassis (NI PXIe-1082) that contained 2.3 GHz Quad-Core Controller (PXIe-8135), 50 MHz digital I/O (NI PXIe-6537), and an expansion module (NI 8262) connected with an external hard drive (NI HDD-8265). The DAQ system can record 32 individual data channels at up to 50 MHz each and is rugged and portable with 6 TB data storage. List-mode data acquisition and analysis software were also designed at LANL for the fresh fuel DDA project.

3. Experimental Data

3.1 Deadtime Correction

Before any analysis is possible, the experimentally measured data must be corrected for the effects of the detector pre-amplifier deadtime. As each detector has its own preamplifier, and due to low overall efficiency, the detected neutron rates by individual detectors are considered to be independent of each other and therefore can be corrected using the infinite exponential method [10]. It is generally well accepted that the dead time of individual preamplifiers is paralyzable and around 1 ms although the exact values differ for individual pre-amplifier models. Furthermore, given the high rate that follows immediately after the interrogating neutron pulse, the dead time correction is very sensitive to the actual value of the correction parameter used. In order to explore the sensitivity of the deadtime correction we utilized three different dead time correction parameters 800, 850, and 900 ns in two different time domains after the interrogating neutron pulse, 70-100 µs and 100-150 µs. While the deadtime correction parameters were chosen around 850 µs following the comparison of counting rate distributions at low and high neutron generator intensities [10] [11], the two time domains were selected to reflect time domains where properties of the DDA signal may be used for extraction of various fuel assembly parameters as originally outlined in [3].

Following the deadtime correction using the three deadtime correction coefficients, the die-away time has been determined as the decay constant of the exponential fit of the neutron detection rate on individual detectors in a given time domain.

As expected, in the early time domain (70-100 µs), the die-away times of detectors closest to the neutron generator (detectors 1 and 2) were most sensitive to the change in the deadtime correction coefficient, with the die-away time changing by 20% for detector 1. Additionally, fuel assemblies with more fissile material (i.e. higher enrichment), were affected more than assemblies with less fissile material. Detectors positioned further from the neutron generator (and therefore experiencing lower count rates) were considerably less affected by variation of the deadtime correction coefficient (Fig. 2A).

In the later time domain (100-150 µs), the detected neutron rates were not as sensitive (0%-3%) to the variation of the deadtime correction coefficient (Fig. 2B). However, as shown in the next section, this difference in the determined die-away time is still statistically significant and needs to be considered when comparing experimental data to simulation results.

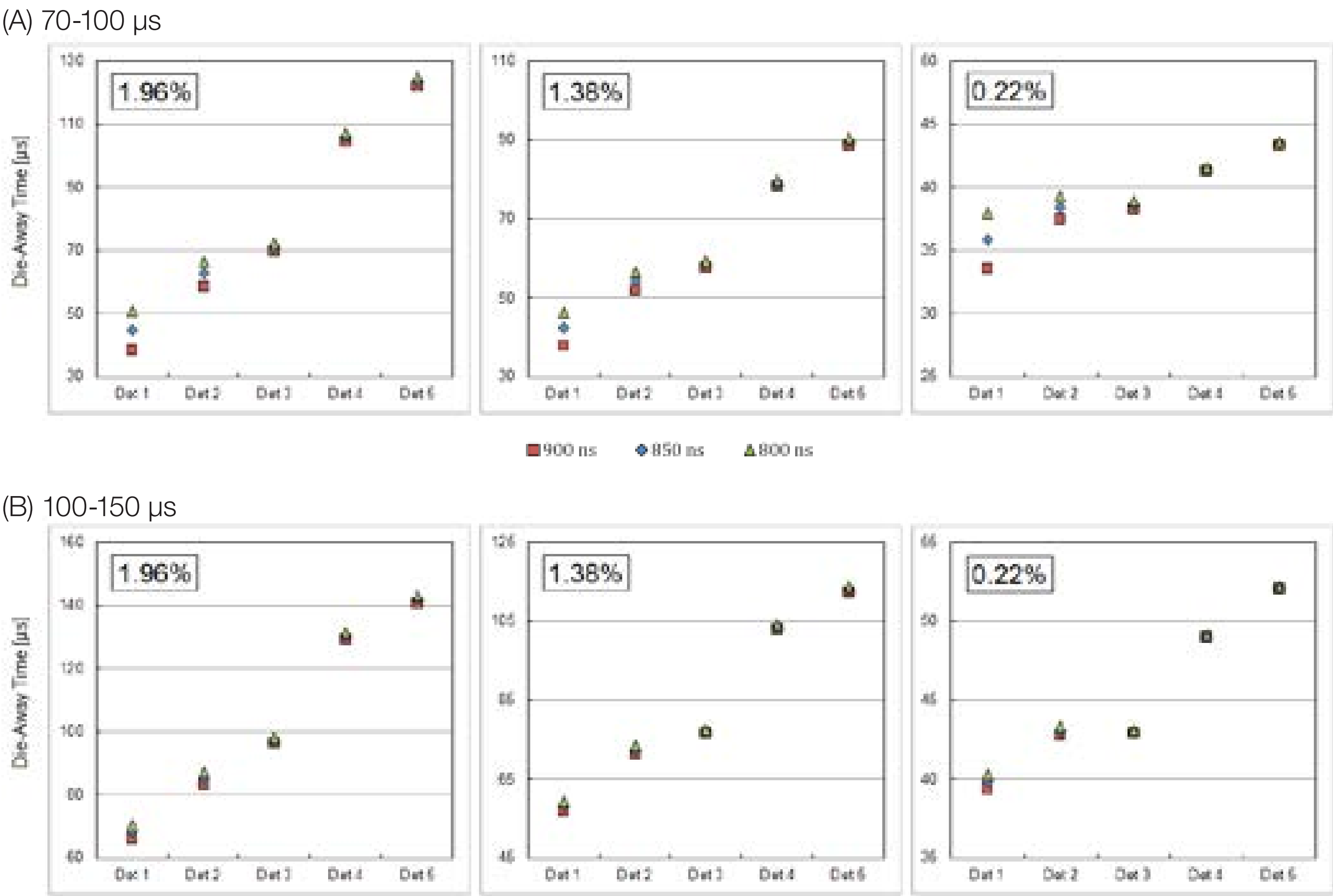


Figure 2: The experimentally measured die-away times in two time domains (A) 70-100 µs and (B) 100-150 µs for three different fresh fuel assembly enrichments (1.96%, 1.38%, and 0.22% ^{235}U) and five different detector positions were calculated using three different deadtime correction coefficients: 800, 850, and 900 ns. The statistical uncertainty of individual values is below 0.6%; therefore, the appropriate error bars would be smaller than individual data symbols.

3.2. Estimation of Die-Away Time Uncertainty

The uncertainty in the experimentally determined die-away times was estimated by recording a series of 10 measurements each 30 s for two fresh fuel assembly (FFA) configurations with different average enrichments (1.67% and 1.09% ^{235}U) and the empty assembly without any fuel pins. For the empty fuel assembly, the DDA signal die-away time depends on the detector system properties, such as the amount of moderating material around the detectors. For each measurement, we determined the die-away time value in the 70-100 µs, 100-150 µs, and 150-200 µs time intervals which were chosen to minimize the dead time effects and maximize the statistical significance of the recorded data. The mean die-away time and the absolute and relative standard deviation (σ) are listed in Table II.

Overall, for the early and mid time domains evaluated, the experimental die-away times generally deviated by less than 1.0 µs, or less than 1.0%, from the mean. In the later time domain, we found larger relative errors due to decreasing count rates. Averaged over all sixty measurements, the standard deviation of individual measurement is approximately 1.0%. Based on these results, and considering typical uncertainties associated with various experimental procedures (calibration, position reproducibility) cited in previous work [12] we concluded that in the case of fresh fuel, a measurement time of 30 s was sufficient for a statistically accurate die-away time determination in the early to mid time (<150 µs) domains. In practice, we expect to greatly exceed this minimum limit and acquire data for upwards of 5-10 min.

Run (30 s)	Die-Away Times [μs]								
	70-100 μs			100-150 μs			150-200 μs		
	1.67%	1.09%	Empty	1.67%	1.09%	Empty	1.67%	1.09%	Empty
1	64.47	51.41	33.69	85.90	66.92	32.36	117.83	101.60	32.90
2	64.93	51.47	33.79	84.27	67.56	32.67	116.50	98.67	32.29
3	64.39	51.90	33.88	85.20	67.57	32.78	114.39	97.71	32.07
4	64.70	51.70	34.33	86.10	67.77	32.35	115.45	98.19	32.16
5	64.25	52.05	34.04	85.16	66.50	32.14	119.92	96.79	32.88
6	63.94	52.50	34.19	87.19	67.32	32.59	117.30	98.93	33.30
7	65.66	52.32	33.17	84.69	67.21	32.14	118.07	96.59	32.17
8	63.75	51.72	33.61	84.81	67.36	32.54	116.19	96.70	32.08
9	63.93	51.11	33.52	85.83	66.29	32.31	116.52	97.33	31.77
10	63.53	52.28	33.89	85.16	67.01	32.43	115.65	98.21	32.46
Mean [μs]	64.36	51.85	33.81	85.43	67.15	32.43	116.78	98.07	32.41
σ [μs]	0.63	0.45	0.34	0.85	0.48	0.21	1.57	1.49	0.48
σ [%]	1.0%	0.9%	1.0%	1.0%	0.7%	0.7%	1.3%	1.5%	1.5%

Table II: The results of ten 30 s measurements: the die-away time in three time domains, the mean value, standard deviation (σ), and relative uncertainty were determined for three fresh fuel cases (1.67%, 1.09%, and 0.49% ^{235}U) and the empty setup.

4. MCNPX Fresh Fuel Simulations

4.1 Simulation Setup

The DDA instrument in the MCNPX simulations was designed to reproduce the experimental setup as accurately as reasonably achievable. All of the main experimental components, including the fresh fuel assembly, water tank, three stainless steel detector pods containing a total of nine ^3He detectors, and the DT neutron generator in a waterproof stainless steel cylinder were modeled, as shown in Fig. 3. The only major instrument component excluded from the simulations is the flux monitor which is currently situated outside of the water tank and thus considered to have negligible influence on the transport of interrogating neutrons. The typical simulation involved computation of $1x10^9$ neutron histories, i.e. neutrons isotropically originating in the target plane of the simulated DT neutron generator with the energy of 14.1 MeV and propagated for 400 or 800 μs. Distributing the computation over a cluster of 17 nodes each with eight 2.43 GHz processors and 16 GB of RAM resulted in a calculation lasting around 6-7 hours (i.e. 800-1000 hours of total computer processing time) with the limiting factors being the cluster availability and size of the memory. Considering these constraints and the intensity of the real life neutron generator, the simulation of $1x10^9$ neutron histories corresponds to only 2-5 seconds of a real life experiment. Material definitions from a Pacific Northwest National Laboratory report [13] were used to standardize the materials in the simulations.

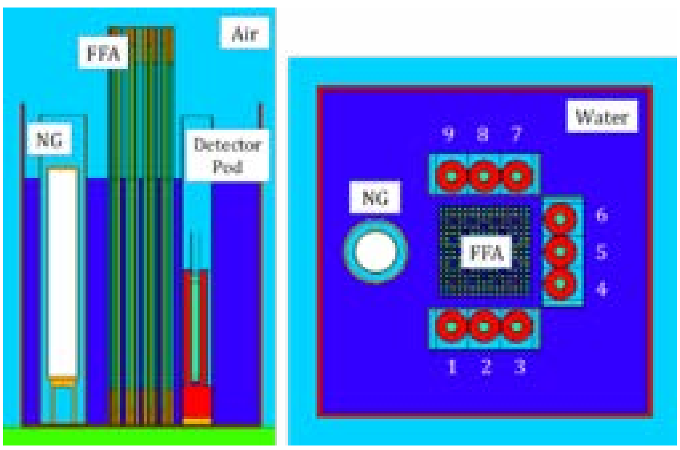


Figure 3: Schematic of the DDA instrument setup as modeled in the MCNPX simulations.

4.2 Sensitivity Studies

Considering the limitations of the simulations in terms of the number of neutron histories that can be simulated in an individual run, the statistical uncertainty (or accuracy) becomes a relevant issue when compared to the experiment that may last significantly longer (10's or 100's of seconds). Therefore we investigated the impact of small variations of several parameters of the MCNPX simulations to determine their impact on the overall results. These sensitivity studies included the evaluation of the statistical variation in MCNPX results, neutron generator pulse wrap-around effects, and small changes to the detector positions.

4.2.1 Statistical Variation in MCNPX

The error of the MCNPX tally results is calculated by the code itself based on analysis of sub-sections of the simulation. In order to estimate the uncertainty on values obtained when the MCNPX results are processed further, such as calculating the die-away time, we performed multiple MCNPX simulations with different random number seeds and evaluated the deviation in the final results.

Five statistically identical simulations, differing only by the random number seed, were performed for two different and extreme cases with the highest and the lowest (i.e. no) fissile content, a 1.96% ^{235}U fresh fuel enrichment and an empty assembly. The die-away times for the nine detectors positioned around the fuel assembly were determined from the DDA signal. The mean value and the standard deviation of the die-away times were found for three different time domains: 70-100 μs, 100-150 μs, and 150-200 μs (Table III) which were selected to simply map the statistical uncertainty of the simulations over a region of potential interest that is inspired by previous work [3] [4].

The deviation in the die-away times has been found to be less than 2.5% for the early to mid time domains when different random number seeds were used, indicating that the statistical quality of each simulation was sufficient with respect to the experimental reproducibility in the early time domains (<150 μs), but may be already limiting in the later time domains (>150 μs). The 1.96% ^{235}U case was more affected by the different random number seeds than the empty fuel assembly case. The back detectors, primarily detector 5, experienced the largest deviation in die-away time. The relative error for all detector positions for both fuel assembly configurations increased with time after the neutron pulse, as can be expected due to the declining population and the decreased number of detected neutrons. Typical uncertainties for the simulated die-away time values ranged from less than 1% to approximately 4% per detector position for the two fuel configurations.

4.2.2 Neutron Generator Pulse Wrap-Around Effects

In each MCNPX simulation, all neutron histories are considered independent of each other and are followed

	1.96% ^{235}U: DDA Signal Die-Away Time [μs]								
	70-100 μs			100-150 μs			150-200 μs		
Run	Det 1	Det 3	Det 5	Det 1	Det 3	Det 5	Det 1	Det 3	Det 5
1	46.5	62.3	93.1	59.7	85.0	136.2	89.3	120.0	176.2
2	47.7	60.7	93.0	60.8	83.8	135.4	90.1	128.7	157.7
3	48.0	60.8	93.4	59.9	87.4	142.3	90.2	129.2	169.2
4	47.2	62.9	97.8	60.2	86.4	137.0	92.7	122.2	164.4
5	47.8	60.9	91.0	60.4	85.3	141.9	90.8	127.0	161.8
Mean [μs]	47.4	61.5	93.7	60.2	85.6	138.6	90.6	125.4	165.8
σ [μs]	0.6	1.0	2.5	0.4	1.4	3.3	1.3	4.1	7.1
σ [%]	1.2%	1.6%	2.7%	0.7%	1.6%	2.4%	1.4%	3.3%	4.3%

	Empty: DDA Signal Die-Away Time [μs]								
	70-100 μs			100-150 μs			150-200 μs		
Run	Det 1	Det 3	Det 5	Det 1	Det 3	Det 5	Det 1	Det 3	Det 5
1	33.7	33.9	35.1	32.7	33.4	32.0	33.0	33.5	31.5
2	34.4	33.8	35.1	33.4	33.1	34.2	32.6	32.1	33.0
3	34.7	34.2	34.2	32.8	34.1	32.4	32.5	32.3	33.6
4	34.0	33.9	33.8	33.5	32.9	33.2	32.4	32.9	33.8
5	34.5	33.7	33.9	33.2	33.7	34.0	32.4	32.0	34.1
Mean [μs]	34.3	33.9	34.4	33.1	33.4	33.2	32.6	32.6	33.2
σ [μs]	0.4	0.2	0.6	0.4	0.5	1.0	0.3	0.6	1.0
σ [%]	1.2%	0.5%	1.9%	1.1%	1.4%	2.9%	0.8%	2.0%	3.1%

Table III: The standard deviation of the mean simulated die-away time in three time domains was calculated from five MCNPX simulations starting with different random numbers to determine the statistical variation of the results. The relative error increased in the later time domains. The back detector (detector 5) was most affected by statistical variation in the transport code.

(i.e. neutron transport is performed) for a prescribed length of time, e.g. 400 or 800 µs. This means that each neutron history represents a small part of a single neutron generator pulse, and that overall essentially only one (though unrealistically strong) neutron generator pulse is simulated when results of all neutron histories are superimposed on one another. During an experiment, the neutron generator pulses tens of thousands of times as the DDA signal is typically acquired over several minutes with each pulse superimposing the tail of the previous pulse. We investigated how neutrons from the previous pulse which may still be present in the fuel assembly and induce fission could change the die-away time observed by the detectors.

During the experiment, the DT neutron generator is typically operated at 2500 Hz with a 20 µs long pulse. Therefore, every 400 µs a new pulse from the generator arrives to interrogate the fuel assembly. To reproduce the wrap-around effects, the tally signal from the 400-800 µs time domain respective time bins were summed with the signal from the 0-400 µs time domain. Simulated data not including the wrap-around neutrons (Fig. 4, "Original") and including the residual neutrons (Fig. 4, "Wrap Around") were plotted. Including the residual neutron population slightly increased the overall number of neutrons detected in the 0-400 µs time domain and slightly increased the DDA signal magnitude and the die-away time. The previous pulse neutron population (Fig. 4, "Previous Pulse") is approximately two orders of magnitude less than the DDA signal from the subsequent pulse but still influences the recorded signal. On average, the pulse wrap around effect increased the DDA die-away time by approximately 2% for all detector positions and fuel configurations (Fig. 5). The empty FFA case was not affected because the ^{3}He detectors wrapped in Cd are not sensitive to thermal neutrons and therefore prompt fission neutrons are required to create a detectable signal. The results in this study have been corrected for the wrap-around effect, which is however not expected to significantly change the uncertainty estimate discussed in the previous sections.

Another potential effect is the influence of delayed neutrons on the DDA signal. Technically, delayed neutrons are included in the MCNPX simulations; however, the time cut-off of the tally for a single neutron history (1 ms) effectively excludes delayed neutrons from contributing to the tally. Based on the previous simulation study [14], we expect delayed neutrons to contribute a gradually increasing background to the DDA signal with a time constant on the order of 30 s that reaches its maximum in about five minutes of neutron generator operation. However, even then the total number of delayed neutrons is expected to be only about 1% of the all neutrons produced by fission. We intend to quantify the influence of delayed neutrons on the DDA signal in a future study.

4.2.3 Detector Position

The effects on the DDA signal magnitude and die-away time by moving the detectors closer and farther from the FFA and moving the detectors horizontally along the side of the FFA were investigated through MCNPX simulations.

Moving the detectors 4 mm away (relative to the best known position) from the fuel assembly resulted in

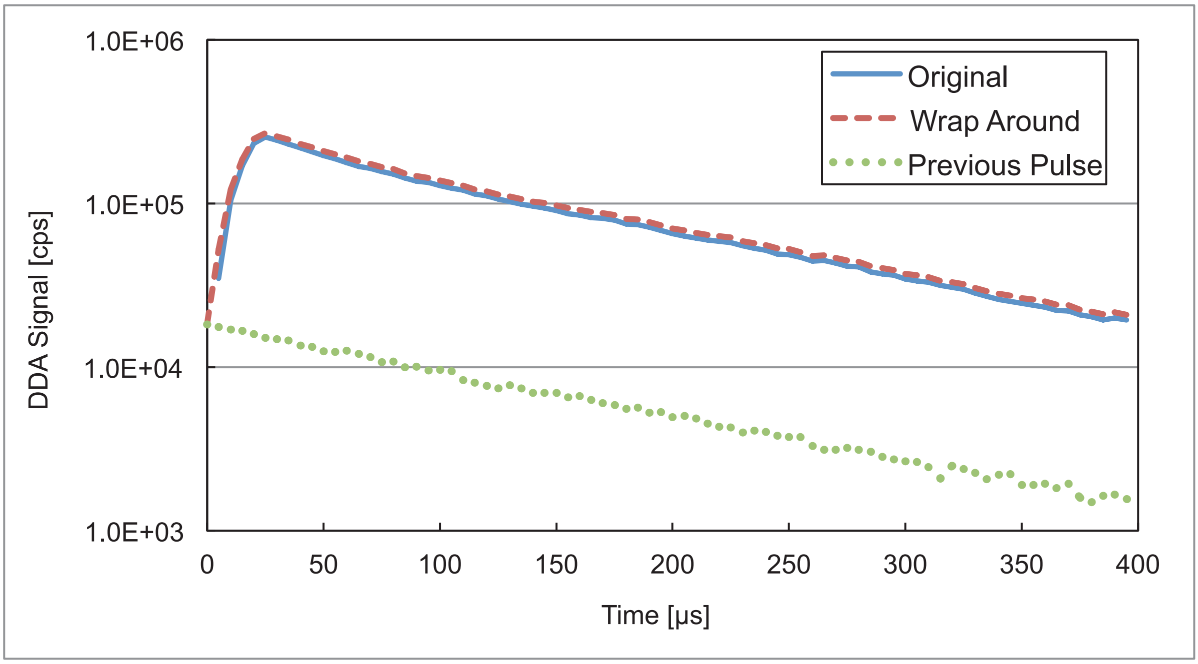


Figure 4: The simulated DDA signal from a single MCNPX simulation ("Original") was compared to the previous pulse wrap around effects included signal ("Wrap Around") from Detector 5. The magnitude of the residual neutron population from the previous pulse ("Previous Pulse") was compared to the DDA signal magnitudes.

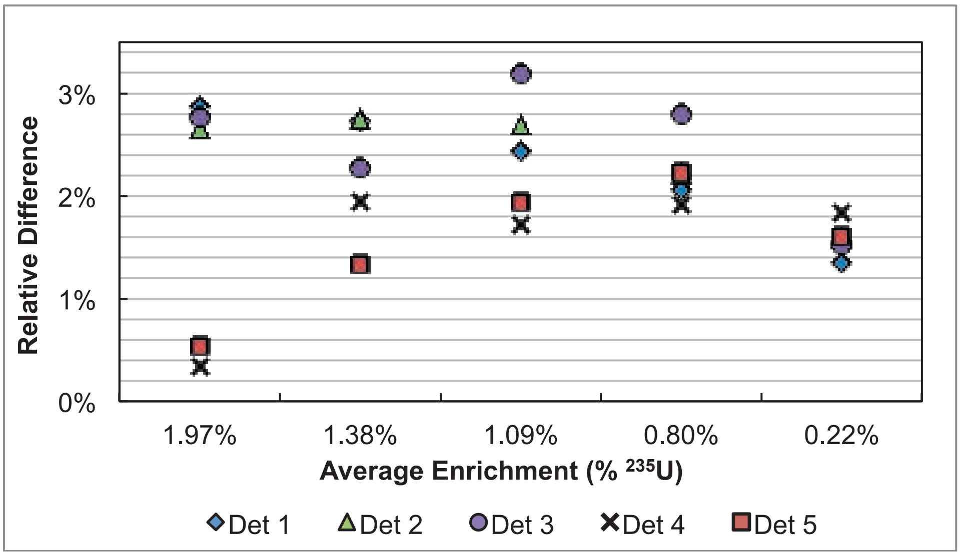


Figure 5: The relative differences between the simulated die-away times in the 100-200 μs time domain of the original and the wrap-around corrected DDA signal. The wrap around corrected die-away time was consistently larger than the original, due to neutrons still present in the vicinity of the fuel. The empty FFA die-away time was not affected by pulse wrap-around effects.

a 1-2% decrease of the DDA signal magnitude recorded from most detectors. Shifting the detectors 4 mm closer to the FFA resulted in a 1-4% increase of the DDA signal recorded by most detectors. Horizontal shifts of the detectors also caused changes to the DDA signal die-away times; however, the magnitude of the effect of the shift was partially dependent on the detector position and time domain over which the die-away time was determined. Moving the detectors ±2 mm horizontally along the side of the FFA from the best known position resulted in the largest changes to the back detector (detectors 4-6) die-away times in the 70-100 μs and 100-150 μs time domains. The back detector die-away times changed approximately 2-6% with the horizontal detector shifts.

In future experiments, we will use additional components to strictly fix the positions of the detectors relative to the FFA as small changes to the DDA instrument geometry results in measurable changes to the DDA signal magnitude and die-away time.

5. Comparison of Experiment and Simulation

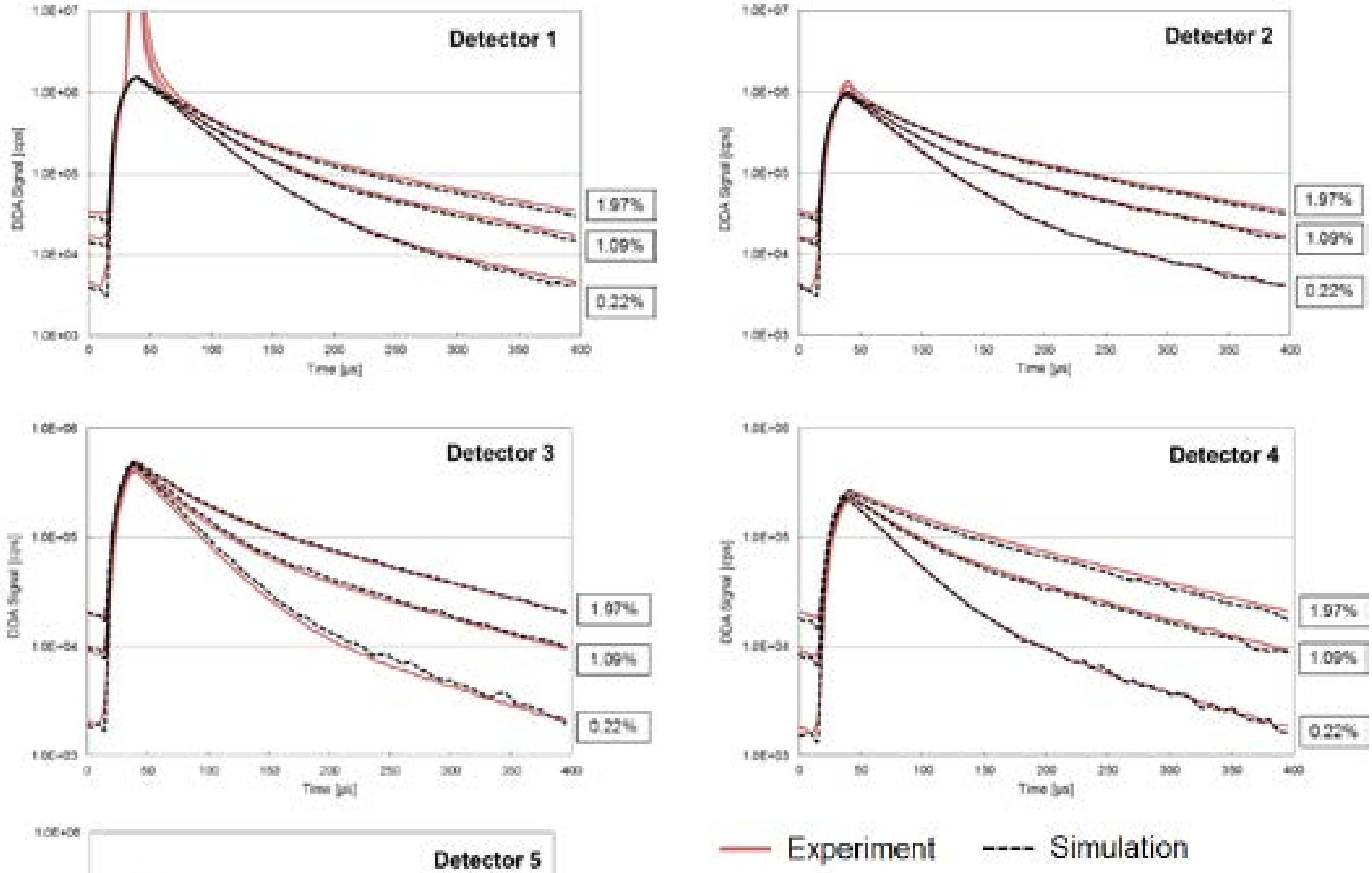
The experimental and simulated results were compared to determine how accurately we are able to model the complex DDA signal using MCNPX. We evaluated two observables from both the experimental data and the simulation results: the time-dependent behavior of the DDA signal and the DDA signal die-away time magnitude in two time domains.

From previously performed niobium foil irradiations, the DT neutron generator yield at the operating high voltage of 90 kV was estimated to be $1.8 \cdot 10^8$ n/s ± 5%. A deadtime correction coefficient of 875 ns was used to correct the experimental data. Both the experimental and simulated data were acquired in 5 μs bins; the DDA signals were then converted to counts per second.

5.1 The Dynamic Evolution of the DDA Signal

The experimental (solid red) time-dependent DDA signal was plotted with the MCNPX simulation results (black dashes) in Figure 6. Overall, we found the experimental and simulated DDA signal distributions compared well within experimental and simulation uncertainties. The DDA signals trended well for multiple enrichments and detector positions.

Detector 1 was heavily impacted by deadtime directly after the neutron generator pulse. The deadtime effects are due to the very high count rate experienced by the detector and electronics due to its position close to the DT neutron generator. The count rates in the front detectors (detectors 1 and 2) were so high that the deadtime correction model failed at the earliest times (<70 μs), giving rise to signal excursions (seen above the experimental pulse peak). In the future, we intend to upgrade our experimental detector/electronics packages by using LANL-made faster post-burst recovery electronics [11] to improve the data quality in the early time domains. We will also consider reducing the efficiency of the front detectors by decreasing the radial thickness of the HDPE sleeves. However, we need to find a balance between the front and back detector efficiency, as we do not want to adversely affect the back detectors which already experience much lower count rates.



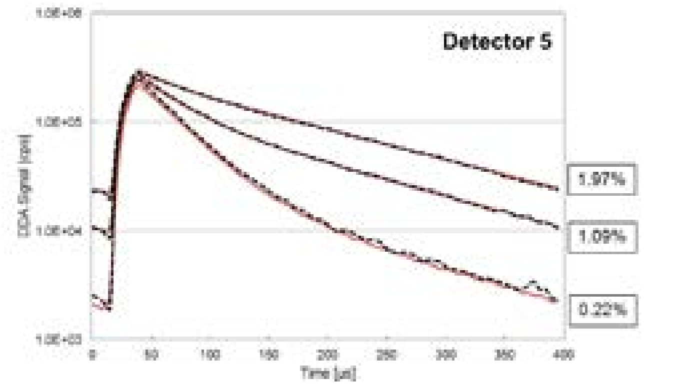


Figure 6: The dynamic evolution of the DDA signal is plotted for three fresh fuel enrichments (1.97%, 1.09%, and 0.22%) for five detectors positioned around the FFA.

5.2 Die-Away Time as a Function of Enrichment

The DDA signal die-away time values for the 70-100 µs and 100-150 µs time periods were determined from the experimental and simulated results and compared. In the early time domain (70-100 µs), the MCNPX simulated die-away times trended well with the experimental die-away times (Fig. 7). Detectors 1 and 2 were particularly sensitive to deadtime correction in the early time domain due to the very high count rates recorded by the front positions (close to the DT neutron generator). The die-away times for detectors 3, 4, and 5 compared well for all fresh fuel enrichments and the empty case. We also found good agreement between the experimental and simulated die-away times in the later time domain (100-150 µs) when deadtime was no longer significantly affecting the DDA signal.

We determined the relative differences between the experimental and simulated DDA signal die-away times for the 70-100 µs and 100-150 µs time domains (Fig. 8). We found good agreement between the experimental and simulated die-away times both in terms of trending with the average enrichment of the fresh fuel assembly and the magnitude. In the early time domain (70-100 µs), the experimental DDA signal from detector 1 was recovering from the DT neutron generator burst which affected the die-away time magnitude. However, the die-away time values from detectors 2, 3, 4, and 5 compared well with the simulated results, with an average relative difference of approximately ±2.3%. In the later time domain (100-150 µs), the die-away time values compared well for all detectors for all fresh fuel arrangements, with an average relative difference of approximately ±2.6%.

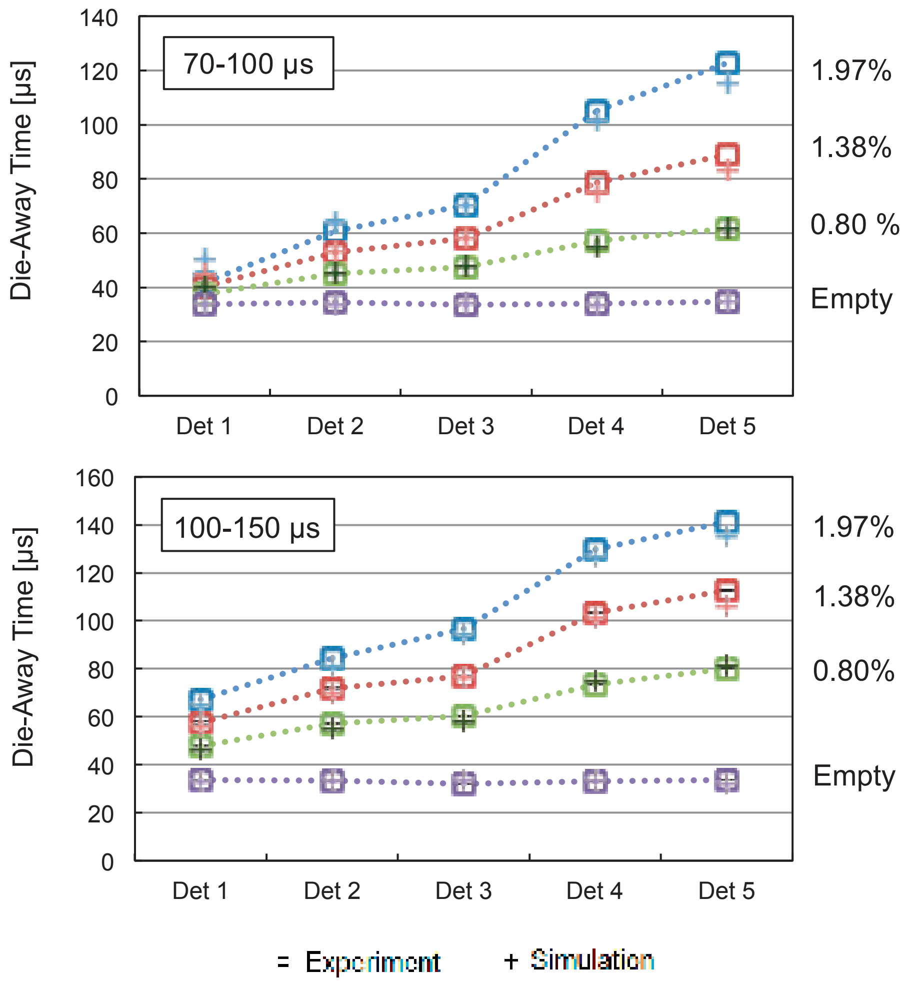


Figure 7: Comparison of the experimental (square) and simulated (cross) die-away times of the DDA signal in the 70-100 µs and 100-150 µs time domain for four FFA enrichments. The statistical uncertainty of individual values is below 0.6%; therefore, the appropriate error bars would be smaller than individual data symbols.

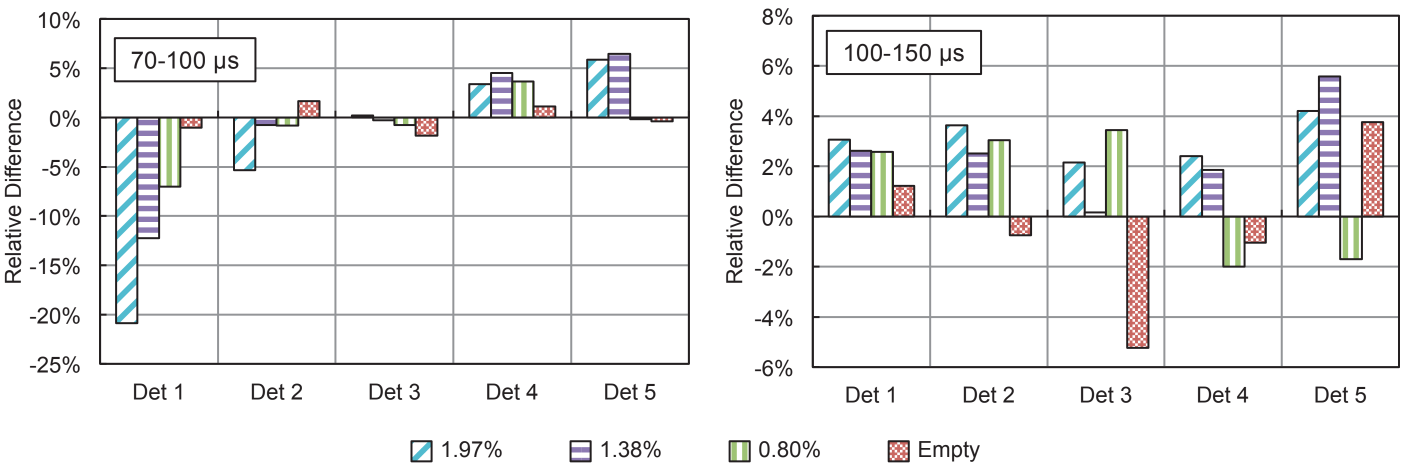


Figure 8: In the early time domain (70-100 µs), the die-away times as a function of average FFA enrichment from Detectors 2, 3, 4, and 5 compares fairly well with the simulated results, with an average relative difference of approximately ±2.3%. Detector 1 was overwhelmed by deadtime which affected the die-away time determination. In the later time domain (100-150 µs), the die-away times as a function of average FFA enrichment from all detectors compared well with the experimental results, with an average relative difference of ±2.6%.

From the mainly positive relative difference values for both time domains, the simulation generally underestimated the DDA signal die-away time. The minor discrepancies between simulation and experiment die-away time magnitudes may have been caused by small geometry errors or the lack of delayed neutrons contributing to the DDA signal in simulation.

6. Conclusions

Overall, we found good agreement between the fresh fuel experiments and the simulations, with the relative difference between the die-away times measured or simulated being within the combined uncertainty of the experiment and the simulations

Considering the significant dynamic range of parameters explored in this study, we have shown that the DDA instrument is capable of practically measuring the complex time-dependent signal from a fresh fuel assembly that is interrogated by an external pulsed neutron source. Through the experiments and simulations described in this paper, we also demonstrate that MCNPX can produce a reliable and sufficiently accurate model of a fuel assembly assay by a DDA instrument.

Using the less complex fresh fuel, this study confirms general trends reflecting the overall physics of the DDA instrument as previously identified through spent fuel simulation which assumes that the dynamic evolution of the DDA signal reflects the effective fissile and neutron absorber content, not a particular isotopic composition. These results are therefore in support of conclusions derived from simulations of spent fuel assay which predict the capabilities of the DDA instrument to characterize spent fuel for nuclear safeguards applications.

7. Acknowledgements

The authors would like to acknowledge the support of the US National Nuclear Security Administration's Office of Defense Nuclear Nonproliferation Research and Development.

8. Legal Matters

We agree that ESARDA may print our names/contacts 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.

9. References

[1] D. Pelowitz and et al., "MCNPX User's Manual, Version 2.7.0," Los Alamos National Laboratory, LA-UR-11-00438, 2011.

[2] M. Humphrey, S. Tobin and K. Veal, "The Next Generation Safeguards Initiative's Spent Fuel Nondestructive Assay Project," *Journal of Nuclear Materials Management,* vol. 40, no. 3, 2012.

[3] V. Henzl, M. Swinhoe, S. Tobin, H. Menlove, J. Galloway and D. Won Lee, "Direct Measurement of Initial Enrichment, Burn-up and Cooling Time of Spent Fuel Assembly with a Differential Die-Away Technique Based Instrument," in *Institute of Nuclear Materials Management Conference Proceedings*, Orlando, 2012 Los Alamos National Laboratory, LA-UR-12-22982.

[4] V. Henzl, "Evaluation of Differential Die-Away Technique Potential in Context of Non-Destructive Assay of Spent Nuclear Fuel," Los Alamos National Laboratory, LA-UR-14-29224, 2014.

[5] "ASTM C1493-09, Standard Test Method for Non-Destructive Assay of Nuclear Material in Waste by Passive and Active Neutron Counting Using a Differential Die-Away System," ASTM International, West Conshohocken, PA, 2009.

[6] K. Coop, "Neutron dieaway methods for criticality safety measurements of fissile waste," in *Winter meeting of the American Nuclear Society*, San Francisco, 1989.

[7] J. Caldwell, R. Hastings, G. Herrera and W. e. a. Kunz, "The Los Alamos Second-Generation System for Passive and Active Neutron Assays of Drum Size Containers," Los Alamos National Laboratory, LA-10774-MS, 1986.

[8] J. Caldwell and W. Kunz, "Experimental Evaluation of the Differential Die-Away Pulse-Neutron Technique for the Fissile Assay of Hot Irradiated Fuel Waste," LA-UR-82-788, Los Alamos National Laboratory, 1982.

[9] AMPTEK, "A111 Charge Sensitive Preamplifier & Discriminator," AMPTEK, [Online]. Available: http://www.amptek.com/products/a111-charge-sensitive-preamplifier/.

[10] A. Goodsell, M. Swinhoe, V. Henzl, K. Ianakiev, M. Iliev, C. Rael and D. Desimone, "Differential Die-Away Instrument: Report on Neutron Detector Recovery Performance and Proposed Improvements," Los Alamos National Laboratory; LA-UR-14-27369, 2014.

[11] K. Ianakiev, M. Iliev and M. Swinhoe, "High Count Rate Thermal Neutron Detectors and Electronics," IAEA-CN-220, 2014.

[12] A. Goodsell, V. Henzl, M. Swinhoe, C. Rael and D. Desimone, "Differential Die-Away Instrument: Report on Comparison of Fuel Assembly Experiments and Simulation," Los Alamos, National Laboratory, LA-UR-15-20242, 2015.

[13] R. McConn, C. Gesh, R. Pagh, R. Rucker and R. Williams III, "Compendium of Material Composition Data for Radiation Transport Modeling," PNNL-15870 Rev. 1, Richland, 2011.

[14] P. Blanc, H. Menlove, S. Tobin, S. Croft and A. Favalli, "An Integrated Delayed-Neutron, Differential-Die Away Instrument to Quantify Plutonium in Spent Nuclear Fuel," *Journal of Nuclear Materials Management,* vol. 40, no. 3, pp. 70-77, 2012.

A Qualitative Analysis of the Neutron Population in Fresh and Spent Fuel Assemblies during Simulated Interrogation using the Differential Die-Away Technique

Niklas Lundkvist[a,b], Alison V. Goodsell[b], Sophie Grape[a], John S. Hendricks[b], Vladimir Henzl[b], Martyn T. Swinhoe[b] and Stephen J. Tobin[a,b,c]

[a]*Uppsala University, S:t Olofsgatan 10B, 753 12 Uppsala, Sweden*
[b]*Los Alamos National Laboratory, P.O. Box 1663, Los Alamos, NM 87545, USA*
[c]*Swedish Nuclear Fuel and Waste Management Company (SKB), Blekholmstorget 30, Box 250, SE-101 24 Stockholm*
E-mail: tobin@lanl.gov,niklaslundkvist@gmail.com

Abstract:

Monte Carlo simulations were performed for the differential die-away (DDA) technique to analyse the time-dependent behaviour of the neutron population in fresh and spent nuclear fuel assemblies as part of the Next Generation Safeguards Initiative Spent Fuel (NGSI-SF) Project. Simulations were performed to investigate both a possibly portable as well as a permanent DDA instrument. Taking advantage of a custom made modification to the MCNPX code, the variation in the neutron population, simultaneously in time and space, was examined. The motivation for this research was to improve the design of the DDA instrument, as it is being considered for possible deployment at the Central Storage of Spent Nuclear Fuel and Encapsulation Plant in Sweden (Clab), as well as to assist in the interpretation of the both simulated and measured signals.

Keywords: differential die-away, spent fuel, used fuel, active neutron interrogation

1. Introduction

The differential die-away (DDA) technique [1] is one of several nondestructive assay techniques investigated as part of the Next Generation Safeguards Initiative Spent Fuel (NGSI-SF) Project [2,3]. The purpose of the NGSI-SF project is to use nondestructive assay (NDA) technologies to strengthen the technical toolkit of safeguard inspectors and others to determine the following technical goals more easily and more accurately:

- Detect replaced or missing pins from spent fuel assemblies (SFA) to confirm item integrity and deter diversion,
- Determine plutonium mass and related plutonium and uranium fissile mass parameters in SFAs, and
- Verify initial enrichment, burnup, and cooling time of facility declaration for SFAs.

The project also includes two related goals with facility-specific benefits: (1) estimation of the heat content and (2) estimation of reactivity (multiplication).

The DDA technique as deployed involves the active interrogation of a fuel assembly with a burst of neutrons. The signature typically used in analysis comes from integrating the counts in a time interval within the dynamically evolving neutron population in the nuclear fuel assembly. The DDA technique can be used to measure the multiplication of a fuel assembly [4], to estimate the plutonium mass [5] or the content of fissile material in the form of the effective ^{239}Pu mass ($^{239}Pu_{eff}$) [6], to estimate the initial enrichment, burnup, and cooling time of the fuel [7] and to detect missing fuel, however, these capabilities are not the primary subject of this paper. Ideally it will be possible to make such determination without using operator declaration; it is part of the NGSI-SF Project research plan to examine the capability of individual nondestructive assay (NDA) techniques as part of the overall NGSI-SF Project research plan of assessing integrated NDA systems in pursuit of the identified goals.

The DDA technique actively interrogates a fuel assembly leading to the release of primarily prompt neutrons from the fission of fissile, and to a lesser degree fertile, material in the spent fuel assembly. A variety of pulsed neutron source can be used for this active interrogation; in the current investigation a deuterium-tritium (DT) neutron generator was selected. As the system is subcritical, the induced neutron population decreases with time with a half-life that is on the order of hundreds of microseconds. The measured DDA signal reveals various properties of the fuel assembly, such as multiplication, which can be used to characterize the fuel assembly. Additionally the DDA signal was previously shown to be a function of the isotopic mixture that results from the initial enrichment, burnup, and cooling time of the fuel. For this current research the MCNPX code [8,9] was used to simulate the time and spatial variation of the neutron population for three different DDA setups.

The presented simulations are unique relative to past research in that the 3-dimentional spatial evolution of the neutron population is displayed in 2-dimensional images in time. This unique description of DDA performance was conducted with both fresh and spent fuel assemblies. The primary goal of this current work is to examine the DDA signal in both a spatial and temporal way in order to suggest possible design changes that may better achieve the technical goals of the NGSI-SF Project, particularly in the context of encapsulation/repository safeguards. Previous DDA simulations, which produced favourable results, [4] used a large mass of metal for spectrum tailoring of the generator neutrons and moderation. The original instrument design [10] used ^{3}He neutron detectors, which in a spent fuel environment, required lead (Pb) shielding in order to protect the ^{3}He tubes from gamma radiation. Additionally the previous work focused only on ^{3}He detectors, while this current work, researched a lightweight design of the DDA instrument, included fission chambers which are less sensitive to gamma radiation, making the thick Pb shielding unnecessary. Fission chambers have approximately 1% the detection efficiency per unit length as compared to ^{3}He tubes for thermal neutrons. Since this inefficiency impacts both the signal and the background equally, and the counting rates are high, it is not expected to impact the DDA instrument performance significantly.

2. DDA instrument design

Four different DDA instrument designs are discussed in this paper. Each is described below and illustrated in Fig. 1 A through D. Common to all the designs is the use of 8 tubes, either ^{3}He or fission chambers. Also in each design, the detector tube is surrounded by a few centimeters of polyethylene that is then surrounded by Cd. The polyethylene is included to moderate, and as such, to increase the detection probability of neutrons that were high enough in energy to penetrate the Cd layer. The Cd layer absorbs essentially all neutrons with energies lower than 0.46 eV and is needed in order to ensure that the DDA detectors detect non-thermal neutrons only. In all cases the assemblies and the DDA instrument are submerged in fresh water. Furthermore, all DDA simulations track only the neutrons produced by the neutron generator during a 10 ms burst as well as the neutrons released in fission chains initiated by these neutrons. The neutrons from fission chains started by neutrons released in spontaneous fission in the fuel are not simulated.

The design illustrated in **Fig. 1A** [10] represents the NGSI-SF Project design at the start of this current research. A key driver in the design depicted in Fig. 1A was the need at that time to integrate DDA with delayed neutron detection. The inclusion of delayed neutron detection drove two key features of the design: (1) thick spectrum tailoring material was included to reduce the neutron energy in order to reduce the fission of ^{238}U because ^{238}U is more than 80% of the fuel by weight and ^{238}U is a strong delayed neutron source per fission (~7 times stronger than ^{239}Pu per fission). (2) As labelled in Fig. 1A, only 6 of the ^{3}He tubes were used to generate the DDA signal because only those tubes were covered in cadmium. The two tubes not surrounded in cadmium have elevated detection efficiency and were used only for delayed neutron detection. The ^{3}He tubes had a 1.89 cm diameter and a 5.08 cm length. The instrument simulated was made to fit a 17 x 17 assembly with a 5 mm water gap between the assembly and the detector walls on all sides.

The design illustrated in **Fig. 1B** represents the first iteration towards the lightweight version of the DDA instrument envisioned for a portable use and varies relative to Fig. 1A in the following ways: The square high density polyethylene blocks were replaced with cylinders for which each cylinder is individually lined with cadmium. The metal spectrum tailoring was replaced with water as a "DDA only" instrument can use a weaker neutron generator and function with a higher energy neutron interrogating spectrum. The intended application of Fig. 1B was fresh fuel measurements for the purpose of simulation validation. Lead shielding was maintained between the fuel and the detector even though it was not needed for gamma attenuation in order to reduce the number of variables changed between the two designs and as a starting point for assessing the impact of the lead. The instrument simulated was made to fit a 15 x 15 assembly with a 5 mm water gap between the assembly and the detector walls.

The design depicted in **Fig. 1C** is very similar to that of Fig. 1B with the exception that the lead shielding was eliminated which allowed the ^{3}He detectors to be moved closer to the fuel. The instrument simulated was also made to fit a 15 x 15 assembly but the water gap between the assembly and the detector walls was doubled to 10 mm.

The design depicted in **Fig. 1D** is modified from that illustrated in Fig. 1C in one significant way. A 1 mm layer of cadmium was added between the neutron generator and the fuel. The water gap was also reduced slightly to allow for a 17 x 17 assembly to be used and fission chambers were simulated instead of ^{3}He. This later change lengthens the die-away time of the individual detectors. The fission chamber tubes had a 2.4 cm diameter, and 5 cm length.

3. The fuel simulated

A range of pressurized water reactor fuel, both fresh and spent, was simulated in the course of this research. Yet, in this study the results for only one fresh and one spent fuel case are depicted. The fresh case was for a 15x15 assembly comprised of ^{235}U to an enrichment of 3.19 wt-% with 204 fresh fuel rods (0.475 cm diameter and 131 cm length) and 21 water-filled guide tubes. This specific case was selected to match planned experiments at Los Alamos National Laboratory (LANL). A "fully burnt" assembly was selected for the SFA simulations because the overwhelming majority of spent fuel in the world is fully burnt. Using the simulation capability in the MCNPX code, a Westinghouse 17x17 with an initial enrichment of 4 wt-% was irradiated to have a burnup level of 45 GWd/tU and then cooled for 5-years; this fuel was part of the NGSI-SF Library 1 [11]. The purpose of simulating fresh fuel in a project that is primarily interested in spent fuel is to make sure that we can benchmark our simulations in an environment that is well characterized.

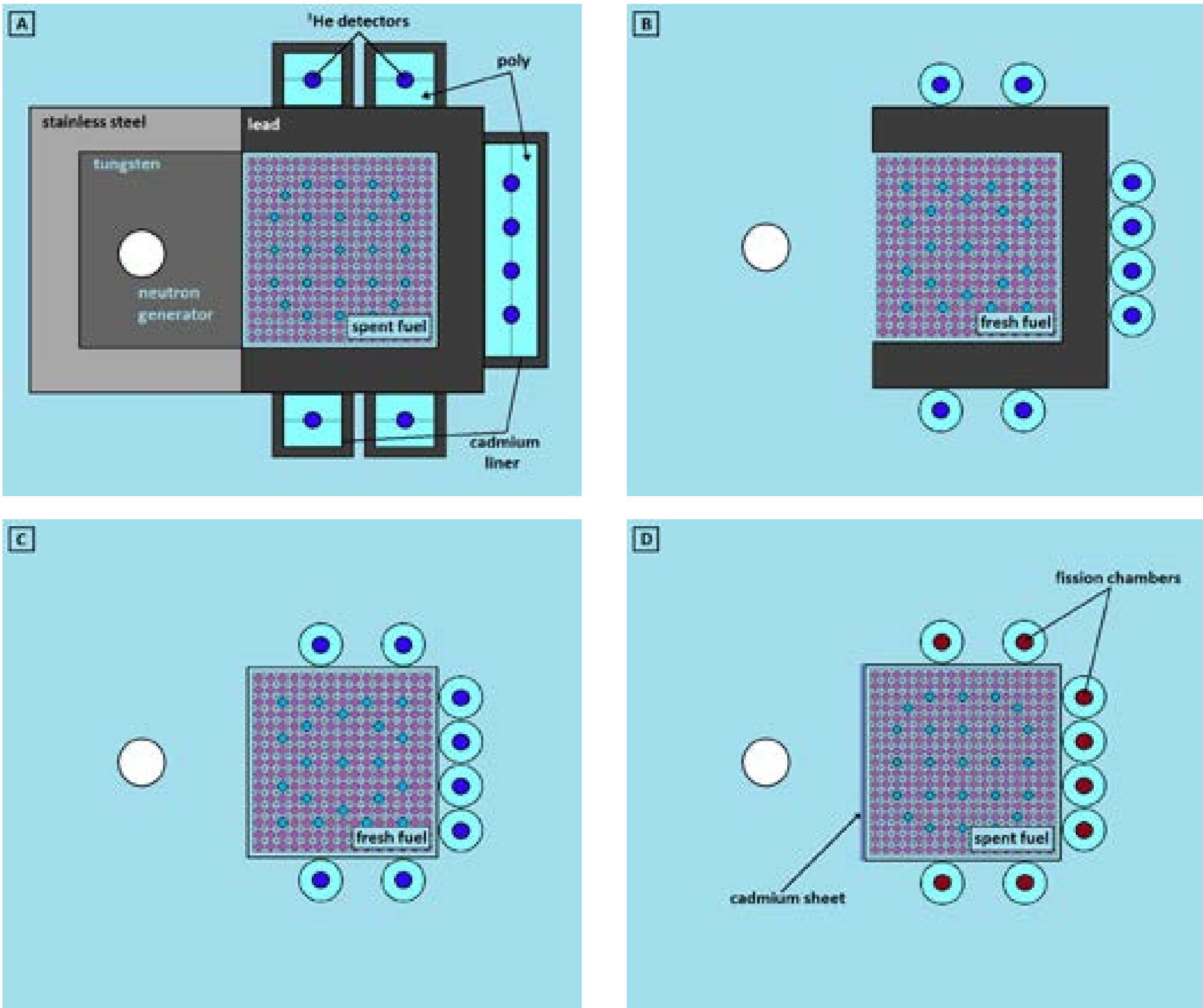


Figure 1: Schematic cross sectional view of the four DDA instrument designs investigated in this study: (A) the spent fuel design used by the NGSI-SF Project at the start of this current research effort, (B) the fresh fuel design that uses water for spectrum tailoring and has individual cadmium lined tubes, (C) the same design as (B) but without the lead shielding which allowed the tubes to move closer to the fuel, and (D) the spent fuel design that is essentially the same as design (C) with the addition of a cadmium sheet and the ^{3}He tubes were replaced by fission chambers.

4. Temporal description of neutron population in fresh fuel

The DDA system simulations were performed using a customized version of MCNPX v273[8,9]. In Fig. 2A and 2B, the number of detected neutrons per neutron generator source neutron for the design depicted in Fig. 1B and Fig. 1C, respectively, are illustrated as a function of time. The "first fission" tallying capability was used to identify if the detected neutron (1) originated in the neutron generator, (2) was part of a chain of events for which the first fission in the chain was ^{238}U, (3) was part of a chain of events for which the first fission in the chain was ^{235}U. The MCNPX capability to identify a detected neutron according to the first induced fission in its chain of interaction was added to the MCNPX code explicitly for this work and is called the "first fission" capability. With the first fission capability, the detected fission contribution from individual isotopes (uranium isotopes for fresh fuel and also plutonium isotopes for spent fuel), and the detected neutrons from the DT neutron generator pulse were tallied to indicate their origin. In these simulations all neutrons start in the neutron generator. If, as a neutron advances through the simulation, it should cause a fission, the identifier or tag on that neutron is changed from originating in the neutron generator to originating in the isotope that underwent fission. From that time onward, the neutron maintains the label it received on its first fission. Analysing these values provides meaningful information on the interaction of the interrogating pulse with the fuel. Additional mesh tallies in the simulation were used to plot the dynamic evolution of the neutron spatial distribution in time for a visual and qualitative understanding of its behaviour.

Basic characteristics of DDA, instrument design 1B and 1C, as applied to spent fuel assemblies are evident in both Fig. 2A and 2B; the detected burst neutrons peak in the time domain before 20 μs. These neutron generator originating neutrons quickly decrease and within 200 μs they are about two orders of magnitude less than the neutrons that originate from ^{235}U actively induced fission. After 50 μs, the neutrons that originate from ^{235}U actively induced fission begin to dominate the detected neutron signal for both the case with the large lead shielding and without it.

There are some differences evident between Fig. 2A and 2B. The detected neutrons that first fission in ^{235}U and ^{238}U both increase in the progression from the design depicted in Fig. 1B to that of 1C; the percentage increase was greater for ^{238}U. The detection efficiency for neutrons created in the assembly increased as we move from the Fig. 1B design to the Fig. 1C design because the ^{3}He detectors moved closer to the fuel and the water layer around the fuel doubled, which increased the multiplication of the assembly. The main point for studying the geometries depicted in Fig. 1B and Fig. 1C was to learn if the illustrated change in geometry had an impact on the ability of the DDA instrument to discern among a range of spent fuel assemblies. It was observed that the temporal variation of the detected neutron signal was a consistent trend with spent fuel variation between the two designs. Hence, leading to the conclusion that both designs are viable designs.

5. Spatial description of neutron population in fresh fuel

In order to understand the spatial qualities of the neutron die-away, the MCNPX code was altered to record a two dimensional depiction of the neutron population as a function of time. This graphical depiction, known in MCNPX terminology as a "mesh tally," was made time-dependent specifically for this NGSI-SF DDA research effort. For the results of this research the neutron population was integrated over voxels that were 7.0 mm by 5.6 mm in the horizontal plane and 310 mm in the vertical plane. In Fig. 3A and 3B, the relative neutron population averaged over two separate time intervals, 10-30 μs and 190-210 μs respectively, for the DDA design illustrated in Fig. 1C are illustrated. The first time interval was selected because that is the interval of time just after the neutron generator is shut off. The later time interval was selected to be consistent with spent fuel simulations illustrated in the next section. Note that the colour scales are not consistent between the two images; yet in both images there is essentially a factor of 3 variation between the pin in which the neutron population is greatest and the pin for which it is least. There is only a slight change in the relative spatial distribution of the neutron population between these two images, which is consistent with the data observed when the spatial images are viewed as a movie. In other words, the neutron population starts and stays relatively centred in the assembly during the measurement duration. From 10 to 70 μs there is only a slight motion of the distribution from the neutron generator side of the assembly toward the centre; from 70 μs out to 470 μs the neutron population remained nearly centred in the assembly. The ending time of 470 μs was chosen because the simulation statistics became poor. Because the neutrons originating in the neutron generator are such a significant part of the detected neutrons signal in these early times, before ~50 ms, the traditional DDA signal, which does not include the interrogating source, can be obtained after ~50 ms when the neutron population is centred in the assembly.

It is important to emphasize that the images in Fig. 3 are for the neutron population. The detected signal can be connected to the neutron population by the detection probability per pin. This is outside the scope of this research. This connection is assembly specific since it

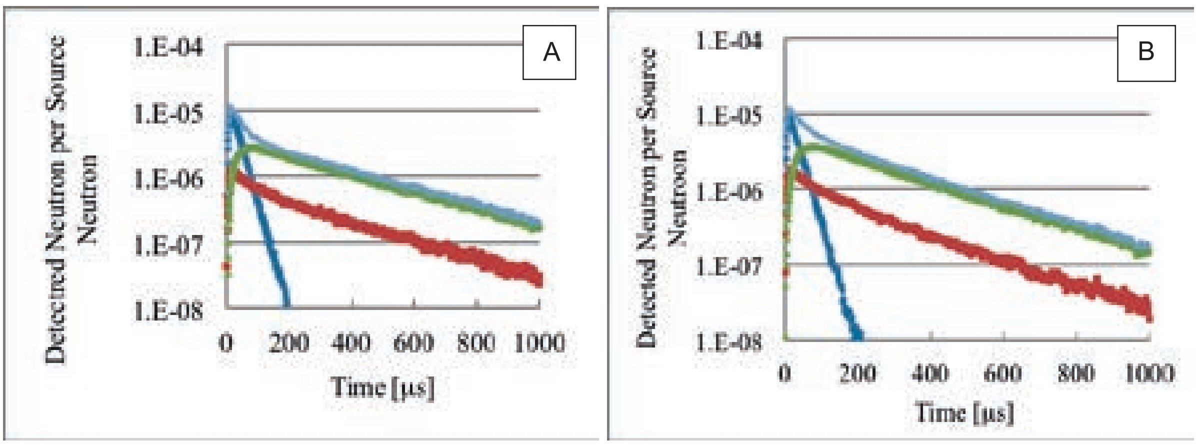


Figure 2: The number of detected neutrons per source neutron for the physical setup illustrated in Fig. 1B and Fig. 1C are depicted for A and B, respectively. The "first fission" tallying capability was used to identify the detected neutrons as originating in the neutron generator (dark blue), ^{238}U (red) or ^{235}U (green); the sum of all detected neutrons are indicated in light blue.

depends on the multiplication of the assembly. For high multiplying assemblies neutron starting in the centre have a greater chance of starting a chain reaction that results in a detected count than do neutrons originating from the centre of a low multiplying assemblies. Although, results for specific assemblies are not presented here, research done with other neutron instruments in the NGSI-SF project have illustrated that in general the detection probability is elevated for neutron originating in the edge of the assemblies particularly for fully burnt assemblies that make up the vast majority of commercial spent fuel assemblies. Hence the combination of the peaked neutron population illustrated in Fig. 3 with the oppositely shaped detection probability distribution is anticipated to make the spatial origins of the detected signal more uniform than the distributions depicted in Fig. 3; yet, the degree to which this "balancing" occurs will be assembly dependent and will be a topic of future research.

The location and density of the neutron population is driven by multiplication, which depends on the fuel composition, as well as the material surrounding the assayed fuel assembly. The population decreases more slowly in regions where the multiplication is the greatest. It is important to note that the fresh fuel case depicted in Fig. 3 is among the more highly multiplying cases expected. This case can be thought of as a bound in terms of the multiplication to be expected from an assembly. The 15 x 15 assembly simulated in Fig. 1C contained rods that were 3.19 wt-% ^{235}U resulting in a net multiplication of ~3.1 for Watt fission spectrum neutrons emitted uniformly from within the assembly with the assembly isolated in fresh water. A typical commercial "fully burnt" 17 x17 assembly, would have a net multiplication of approximately ~2.1, while "fully burnt" smaller assemblies would have still lower multiplication values. This case is presented here for three reasons: (1) it is a bounding case given its high multiplication being similar to a one cycle irradiated assembly that started with ~4.2% ^{235}U initial enrichment, (2) it was anticipated to be one of the measurement setups used at LANL and (3) it provides context for the interpretation of spent fuel results presented later in this study.

6. Spent fuel simulation results

The simulation of fresh fuel measurements was the focus of this research to this point because the research effort is preparing for fresh fuel benchmarking measurements. Yet, because the ultimate goal of the NGSI-SF Project is spent fuel, in this section we present several simulations of spent nuclear fuel assay with the lightweight DDA design. In Fig. 4 A to D the relative neutron population integrated over four different 20 microseconds intervals is illustrated for a 4% wt. initial enrichment, 45 GWd/tU burnup, 5-year cooled assembly; this assembly was chosen since it has roughly the same multiplication as the vast majority of "fully burnt" 17x17 assemblies. The DDA design simulated is the one illustrated in Fig. 1C however with a 17x17 spent fuel assembly. In changing from the 15x15 fresh fuel design used for Fig. 1 and 2 to the 17x17 spent fuel case the assembly width was 1.6 mm larger than the width of the 15x15 assembly, so it was decided to leave the structure of the instrument the same and to allow the water gap around the assembly to be reduced slightly to accommodate this slightly larger assembly.

In contrast to the fresh fuel simulations, the spatial peak in the neutron population in Fig. 4 exhibits a more dynamic behaviour. It appears to migrate into and then out of the spent fuel assembly. The "apparent movement" of the neutron population indicates that the neutron population lives longer in certain regions of the assembly than it does in the other

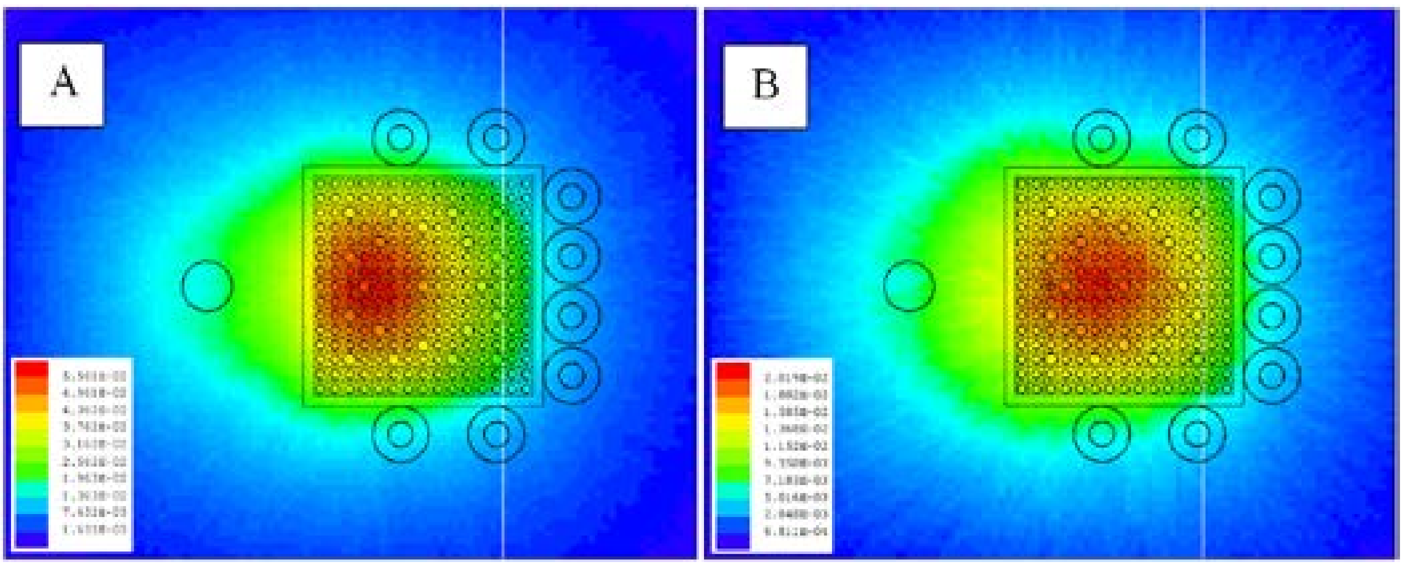


Figure 3: Spatially relative neutron population integrated over 20 µs intervals for the fresh fuel case and DDA design illustrated in Fig. 1C for the two time intervals: (A) 10-30 µs, (B) 190-210 µs (Note: units are in neutrons per source particle over a time interval, but the colour scales are not consistent between the two plots.).

regions. In particular the neutrons live longer in the boundary region at the edge of the assembly (i.e. in the water/fuel boundary) on the neutron generator side of the fuel as compared to the neutron population in either the centre of the assembly or in the boundary region between the fuel and the Cd lined detectors. The population changes among these three regions because the combined effects of the materials in each region foster future generations more or less. The boundary region on the neutron generator side of the assembly produces a larger multiplication in that region as compared to the region in the centre of the assembly or the boundary region on the side opposite to the neutron generator (NG). The boundary region on the three sides of the SFA is especially impacted by the presence of Cd around the detector tubes, which reduces the multiplication in these water/fuel interface areas relative to the water filled neutron generator side of the assembly. In other words, the presence of a relatively thick water region on the left side of the assembly depicted in Fig. 4, for which the absorption cross section is low, elevates the multiplication in the left edge region of the assembly.

The multiplication of the fuel itself changed from 3.1 to 2.1 between the fresh fuel case depicted in Fig. 3 and this spent fuel case depicted in Fig. 4. With the multiplication change the neutron population went from one that was nearly stationary in time and space to a population that varies in time and space. It is important to note that the difference in isotopic composition between the two cases is vast with the fresh fuel case only containing ^{235}U, ^{238}U and ^{16}O while the spent fuel case has a large mixture of actinides and fission products each with its own unique cross sections (absorption and/or fission). The composition of the fuel also changed from the uniform pin-to-pin case of fresh fuel to the non-uniform case of an irradiated assembly. The non-uniformity, in this particular case, is the same in all directions outward from the centre of the assembly because in the calculation of the burn-up of the assembly the model assembly had infinitely reflecting boundary conditions. Hence the non-uniformity of the pin-to-pin variation does not explain why the neutron population appeared to move from near the centre to the left.

From the data illustrated in Fig. 3 and Fig. 4, cases for which the only major change was the fuel itself, we can conclude that the location of the peak in the neutron population in an assembly depends on the assembly itself. The two cases examined illustrate that even in the same environment around the fuel assembly the apparent evolution of the neutron

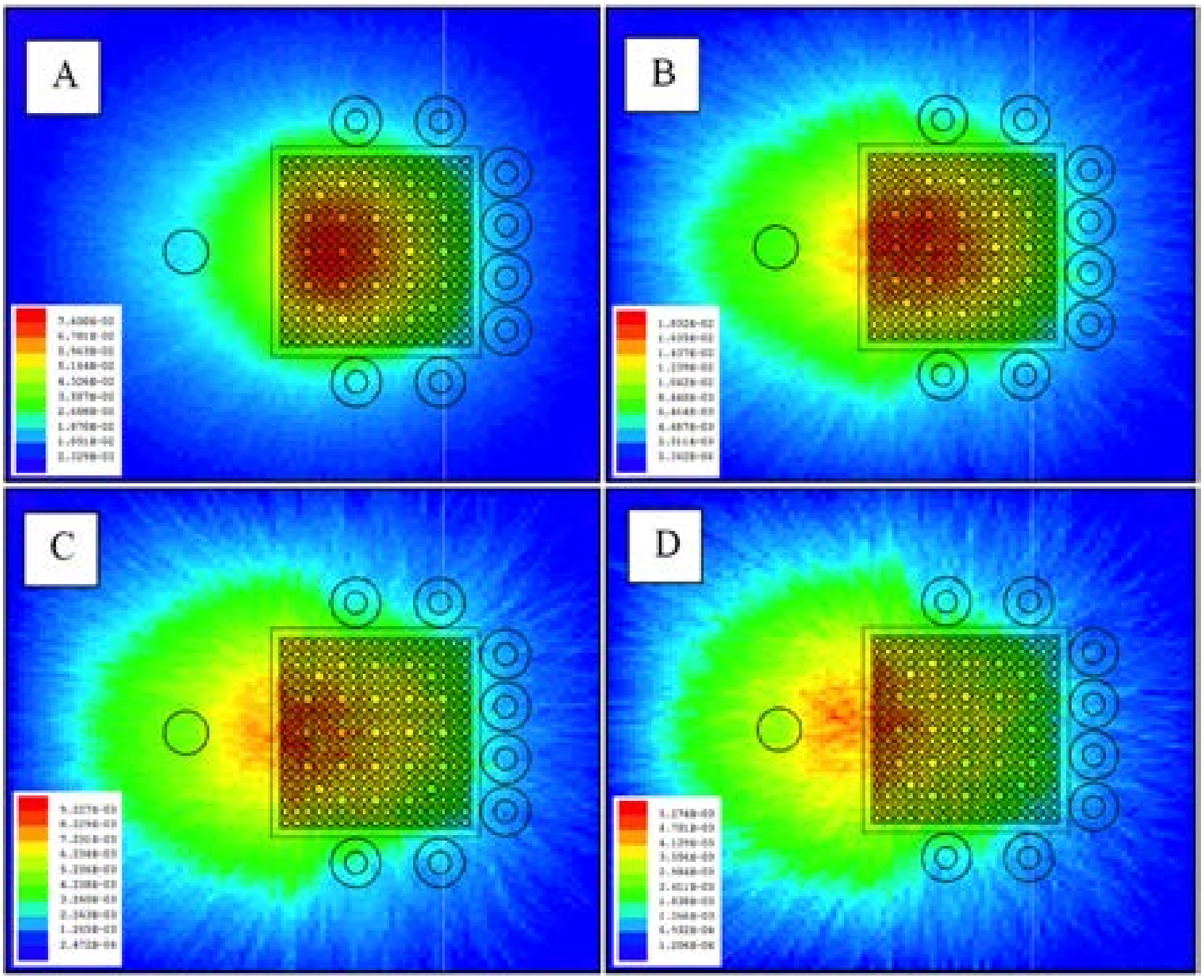


Figure 4: Spatially relative neutron population integrated over 20 µs intervals for the spent fuel case and DDA design illustrated in Fig. 1C for time intervals (A) 10-30 µs, (B) 110-130 µs, (C) 190-210 µs, and (D) 290-310 µs (Note: The color scales are not consistent between the four plots.)

population may be dramatically different both in time as well as space. The neutron population density in the 20 microseconds interval after the neutron generator was turned off is about the same in fresh and spent fuel (Figs. 3A and 4A, respectively). However, during the later time domain of 190-210 microseconds, the density of neutron population drops significantly in the spent fuel, while in the fresh fuel the decrease in the centre of the fuel assembly is only modest. The neutron population dies away much faster in the spent fuel than it does in the fresh fuel because of the reduced multiplication. Additionally, the neutron population on the NG side of the assembly is even larger in the fresh fuel case than in the same region of the spent fuel case. Yet in relative terms, this subpopulation is smaller in the fresh fuel case, and almost dominating in the spent fuel case. As freely as the neutrons can move from the fuel assembly into the water, they can also drift freely back into the fuel assembly. The probability of such re-entry per neutron is identical in both cases studied. While this influx of the thermal neutrons appears to be less significant in the case of fresh fuel, where a large neutron population still survives in the middle of the fuel assembly, it causes a large effect in case of the spent fuel.

From a different perspective, the thermal neutrons that re-enter the fuel from water act as an additional interrogating neutron source. But unlike the precisely controlled neutron generator, the overall intensity and time profile of this additional "neutron source" is not known, i.e. un-normalizable, and depends on the properties of the fuel assembly to be assayed and the boundary material. Moreover, should the SFA not have a constant burnup across all pins, this additional interrogation of the border region by the NG side could result in more or less severe oversampling of part of the SFA that may not be representative of its global characteristics. While this additional "interrogating capacity" can be relatively marginal and thus tolerated (fresh fuel), its effects may be overwhelming in other instances (spent fuel) and should thus be suppressed.

The solution for suppression of the return of the thermal neutrons is evident from the very pictures where this effect can be observed, (i.e. Fig.4). While the neutron population external to the SFA exists on the side of the NG, it is almost entirely absent on the other three sides of the instrument. This is not surprising given the extremely high absorption cross section of cadmium that lines each of the detectors present in these regions, and which effectively prevents any neutron subpopulation build-up. It is therefore suggested that to prevent neutrons thermalized in the water by the NG side of the SFA from re-entering the assembly, a thin Cd sheet be installed along the SFA side. Such cadmium sheet will absorb virtually all neutrons below 0.46 eV, hence virtually all thermal neutrons are prevented from returning to the fuel after scattering in the water.

As a proof of concept on the impact of cadmium, a simple modification was made to the cases simulated in Fig. 4. A 1 mm thick sheet of cadmium was added 10 mm away from the fuel between the neutron generator and the fuel as illustrated in Fig. 1D. This sheet was square and covered the entire side of the assembly. The neutron population for the same 4 time intervals depicted in Fig. 4 is illustrated in Fig. 5 for the same spent fuel assembly.

It is noteworthy that Fig. 4A and Fig. 5A, for the 10 to 30 μs time interval, are nearly identical in terms of the neutron spatial distribution. This is pointed out because the neutrons evolve very differently after this interval in time for the two cases. In the case with the cadmium liner the neutron population moves away slightly from the generator into a central position in the assembly and then stays there. Unfortunately, the neutron population distribution in the fuel in Fig. 5D is not clear because the neutron population located near the generator is so strong. It is worth pointing out that the neutrons outside the Cd liner depicted most prominently in Fig. 5D cannot be detected and cannot re-enter the fuel. The additional sheet of Cd, together with the combined effects of the Cd liners around individual fission chambers effectively isolated the SFA from its surrounding, thus preventing the secondary and undesired interrogation by neutrons reflected back into the fuel. The neutron population inside the fuel better reflects the properties of the fuel itself and spatially stabilizes at the place of the highest multiplication, which in Fig. 5 corresponds to the centre of the SFA.

An additional noteworthy comparison between Figures 4 and 5 involved comparing the neutron population in the 290-310 μs time interval in each figure. In Figures 4 the intensity of the maximum neutron population during this 20 microseconds time interval is difference in both intensity and location relative to Figure 5. The intensity of the maximum population when there is no Cd liner around the fuel, as in Figure 4, is double that which exists when the Cd liner is present, as in Figures 5; indicating that the over all neutron population is reduced by the presence of the Cd liner. Furthermore, when the Cd liner is not present, the maximum population is located on the water-to-fuel interface indicating that this is the optimal region for the neutron population to endure in time. However, when a neutron liner passes through this water-to-fuel interface, the neutron population is severely reduced as indicated in Figures 5; from the colour scale in each figures a factor of 3 reduction of the neutron population, from around 5×10^{-3} to around 1.5×10^{-3} neutrons per source particle.

To further give context to this discussion, it is worth pointing out that a modification in the understanding of how DDA functions is suggested by the 2D images presented in this study especially together with the recent results on the DDA response to assay of asymmetrically burned SFAs [12]. Along with the delayed neutron constraint previously mentioned, the DDA design illustrated in Fig. 1A was designed with a mental expectation that there would be a fission gradient across the assembly at all times during the

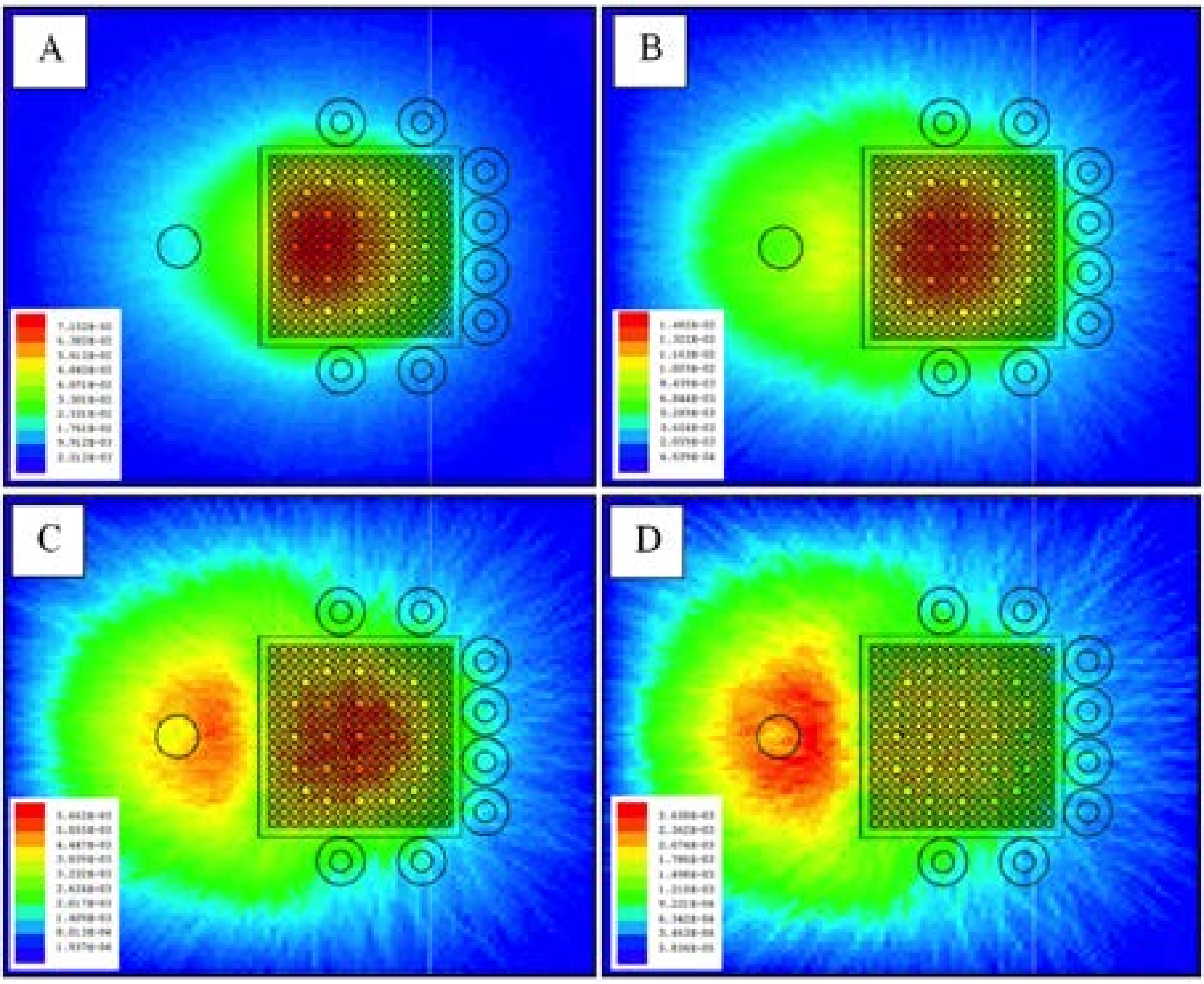


Figure 5: Spatially relative neutron population integrated over 20 μs intervals for the spent fuel case and DDA design illustrated in Fig. 1D for time intervals (A) 10-30 μs, (B) 110-130 μs, (C) 190-210 μs, and (D) 290-310 μs. The presence of Cd in the graph is not visible as it is only 1 mm thick. (Note: The colour scales are not consistent between the four plots.)

assay; the expectation was that the neutron source of the generator would make the neutron population highest on the neutron generator side of the assembly throughout the assay. The logic was that this elevated population near the generator results in elevated detection rates in nearby detectors and thus should be balanced by an oppositely varying efficiency gradient; hence, the detection efficiency should be greatest on the side far from the neutron generator and less near the neutron generator. Separate research has however indicated that different count rates each detector sees may be influenced by local properties of the SFA in its vicinity and that using a sum of all detectors can be inappropriate for characterization of average properties of the SFA [12]. Additionally the thermal return of neutrons from the water or other materials also influences detectors individually as it may cause the neutron population in the SFA to move at a time scale relative to the measurements. That motion can be influenced by both the material composition of the SFA as well as the material surrounding it. Therefore a couple design changes are suggested so that the behaviour of the neutron population provide information about the assembly composition rather than the material around it: (1) a Cd layer around the entire assembly can isolate the assembly from its surroundings if necessary, and (2) detectors should be placed in multiple locations on three sides of the assembly to provide as complete of coverage as possible.

7. Conclusions

In this research three new DDA designs were compared to the NGSI-SF Project design that existed at the start of the research effort. The primary objective of this research is improving the design of the DDA instrument for possible deployment at the Swedish encapsulation facility. A new capability to visualize how the neutron population changes in space and time was utilized and has altered the conceptual understanding of how the neutron population evolves in the assembly. The existence or absence of a large lead layer around the assembly was determined to not significantly impact the performance of DDA for the cases studied. Additionally, one highly multiplying and one multiplying at the level of a fully burnt 17x17 assembly were simulated. The neutron population in both assemblies started in the 10 to 30 μs interval at a location a few rows off-centre. Given this as a starting point, it was shown that the centre of the neutron population can move in time. For the highly multiplying case it stayed nearly centred and for the lower multiplying case it moved to the edge of the assembly near the neutron generator apparently caused by the thermal neutrons returning from water into the fuel. Then it was illustrated that for this later case, the fuel assembly can be effectively isolated from its surrounding environment resulting in the neutron population reflecting the fuel composition rather than coupling it to its environment. The population could be manipulated to stay near the SFA centre by changing the edge conditions with Cd on that side of the assembly to which it had previously moved. The results presented here show that the location of the neutron population inside the assembly can essentially be independent of its surroundings for time intervals after ~50 microseconds. The alteration of the local multiplication in the edge changes the distribution of the neutron population within the assembly. Information learned about the neutron populations in the fresh and spent fuel assemblies from these simulations will be used to improve the designs of DDA instruments deployed in the near future, although varying circumstances merit different designs. The results presented here suggest that the application of a Cd liner is preferable when there is a significant amount of neutron reflecting material present, such as water, directly by or around the fuel assembly. Overall, the work benefitted the NGSI-SF project by understanding better the design challenges of an instrument that can determine the amount of fissile material present in a SFA, help detect partial defects and/or determine the amount of plutonium in the assembly, and to verify properties inherent to the assembly (i.e. initial enrichment, burnup, and cooling time).

8. Acknowledgements

The authors would like to acknowledge the support of the Office of Nonproliferation and Verification Research and Development (NA-22), the Next Generation Safeguards Initiative (NGSI), Office of Nonproliferation and Arms Control (NPAC), National Nuclear Security Administration (NNSA).

9. References

[1] Caldwell W.E. and Kunz, J. T.; **“**Experimental Evaluation of the Differential Die-Away Pulsed-Neutron Technique for the Fissile Assay of Hot Irradiated Fuel Waste,” Richland, Washington, ANS Topical Meeting on Treatment and Handling of Radioactive Waste, 1982.

[2] Humphrey, M.A.; Tobin, S.T., and Veal, K.D.; “The Next Generation Safeguards Initiative’s Spent Fuel Nondestructive Assay Project,” Journal of Nuclear Material Management, Vol. 40, **3**:XL (2012).

[3] Tobin, S.J., Menlove, H.O., Swinhoe M.T. and Schear, M.A.; “Next Generation Safeguards Initiative Research to Determine the Pu Mass in Spent Fuel Assemblies: Purpose, Approach, Constraints, Implementation, and Calibration,” Nuclear Instruments and Methods in Physics Research A, vol. 652, no. 1, pp. 73-75, 2011.

[4] Henzl, V., Swinhoe M.T., Tobin, S.J., Menlove, H.O.; “Measurement of the Multiplication of a Spent Fuel Assembly with the Differential Die-Away Method within the Scope of the Next Generation Safeguards Initiative Spent Fuel Project,” Journal of Nuclear Material Management **3:**XL,p 61-69, (2012).

[5] Henzl, V., Croft, S., Swinhoe, M.T., Tobin, S.J.; “Determination of Pu Content in a Spent Fuel Assembly by Measuring Passive Total Neutron Count Rate and Multiplication with the Differential Die-Away Instrument,” Orlando, FL: Proceedings of Institute of Nuclear Material Management, 2012.

[6] Lee, T.H., Tobin, S.J., Menlove, H.O., Swinhoe, M.T.; “Determining the Pu Mass in LEU Spent Fuel Assemblies: Focus on Differential Die-Away Technique,” Los Alamos National Laboratory report LA-UR-11-00747 (2011).

[7] Henzl, V., Swinhoe, M.T., and Tobin S.J.; “Determination of Initial Enrichment, Burnup, Cooling time of Spent Fuel Assemblies with a Differential Die-Away Techniques Based Instrument,” Orlando, FL: Proceedings of Institute of Nuclear Material Management, 2012.

[8] Hendricks, J.S., McKinney, G.W., Wilcox, T.A., James, M.R.; “MCNPX 2.7.3 Extensions,” Los Alamos National Laboratory Report LA-UR-12-00135 (June 2012).

[9] Hendricks, J.S. and Tobin, S.J.; “NGSI MCNPX Extensions to MCNPX 270,” Los Alamos National Laboratory Report LA-UR-12-00133 (January 2012).

[10] Blanc, P.C., Menlove, H.O., Tobin, S.J., Croft, S., Favalli, A.; “An Integrated Delayed-Neutron, Differential-Die Away Instrument to Quantify Plutonium in Spent Nuclear Fuel,” Journal of Nuclear Materials Management, vol. 40, no. 3, pp. 70-77, 2012.

[11] Trellue, H.R., Galloway, J. D., Fischer, N. A. and Tobin S. J.; “Advances in Spent Fuel Libraries,” Proceedings of Institute of Nuclear Materials Management conference, Palm Desert, California (July 14-18, 2013) and Los Alamos National Laboratory report LA-UR-13-24074.

[12] Martinik, T., Henzl, V., Grape S., Jacobsson Svärd, S., Jansson, P, Swinhoe, M. T., Tobin, S. J.; “Simulation of Differential Die-Away Instruments Response to Active Interrogation of Asymmetrically Burned Spent Nuclear Fuel,” Nuclear Instruments and Methods A, vol. 788, no. 11, July 2015.

Improving the prediction model for Cherenkov light generation by irradiated nuclear fuel assemblies in wet storage for enhanced partial-defect verification capability

Erik Branger, Sophie Grape, Peter Jansson, Staffan Jacobsson Svärd

Uppsala University, Uppsala, Sweden
E-mail: erik.branger@physics.uu.se

Abstract:

The Digital Cherenkov Viewing Device (DCVD) is one of the tools available to an inspector performing verification of the irradiated nuclear fuel inventory in wet storages at a nuclear facility. For gross defect verification, the presence of Cherenkov light and its qualitative properties are sufficient to verify the presence of an irradiated fuel assembly. For partial defect verification, the measured Cherenkov light intensity is quantitatively related to the intensity that is expected from the assembly under investigation, given the operator declarations for that assembly.

While the currently used method for predicting the Cherenkov light emission intensity has performed well, data have also shown that enhanced methods incorporating more details may improve the prediction capabilities even further, in particular for short-cooled fuel assemblies. Fuel parameters such as initial enrichment, burnup, cooling time, as well as the fuel irradiation history and fuel type affect the total emitted Cherenkov light intensity, and should be taken into account in the prediction process. Furthermore, a larger number of fuel types and geometries need to be incorporated into the methods to take geometric effects into account.

This paper describes a new and fast method to predict the Cherenkov light intensity of an irradiated fuel assembly, taking the fuel irradiation history and fuel geometry into account. The proposed method takes advantage of pre-computed Monte Carlo simulations of the Cherenkov light generated by a fuel, and is fast enough to be used in the field. The improved prediction method will also allow for more stringent detection limits, which may improve the partial defect detection capabilities of the DCVD.

Keywords: DCVD; partial defect verification; Cherenkov light

1. Introduction

One of many safeguards tasks undertaken by authority inspectors is the verification of irradiated nuclear fuel assemblies. The fuel assemblies are often stored in water for decay heat removal and for radiation protection. The electromagnetic radiation emitted from the fuel assemblies will interact with the water and gives rise to Cherenkov light in the water, which can be measured. A commonly used method for verifying irradiated nuclear fuel assemblies is to detect and quantify the Cherenkov light emission from irradiated fuel assemblies, and compare the quality and/or intensity of the detected light to what is expected from a fuel assembly.

To do quantitative measurements with the DCVD, the measured Cherenkov light intensity of a set of fuel assemblies are compared to predicted intensities. The currently used prediction method (referred to as CPM in this text) works by first simulating the Cherenkov light intensity in a fuel assembly for a range of burnups and cooling times of the fuel. Based on the operator declared values for burnup and cooling time, the relative intensity of a fuel assembly is then interpolated from the previously simulated data. This paper presents a next generation prediction method (referred to as NGM in this text) for the Cherenkov light intensity, which takes both fuel geometry and irradiation history into account when predicting the intensity.

1.1 Measurements with the DCVD

One of the instruments available to an inspector for measuring the Cherenkov light emitted from a fuel assembly is the Digital Cherenkov Viewing Device (DCVD) [1]. The DCVD has for a long time been used for gross defect verification, where the inspector verifies that an object is an irradiated fuel assembly as opposed to a non-fuel object. The DCVD can also be used to perform partial defect verification, with the purpose of detecting missing and/or substituted fuel rods according to the IAEA's definition of a partial defect, currently at a level of 50%.

In a measurement situation, the DCVD is mounted on the railing of a bridge above the fuel, looking down into the water, as shown in Figure 1. This setup allows for quick, non-destructive measurements of fuel inventories.

Depending on whether an inspector wants to perform a gross or a partial defect verification campaign, the measurement scenario looks slightly different. For gross defect verification, the inspector studies the detected Cherenkov light intensity and light characteristics from each fuel item separately, in order to determine whether the object under study is a fuel item or a non-fuel item. For partial defect verification a collection of irradiated fuel assemblies of the same type is measured, and the measured (quantified) intensities are compared to expected intensities, which have

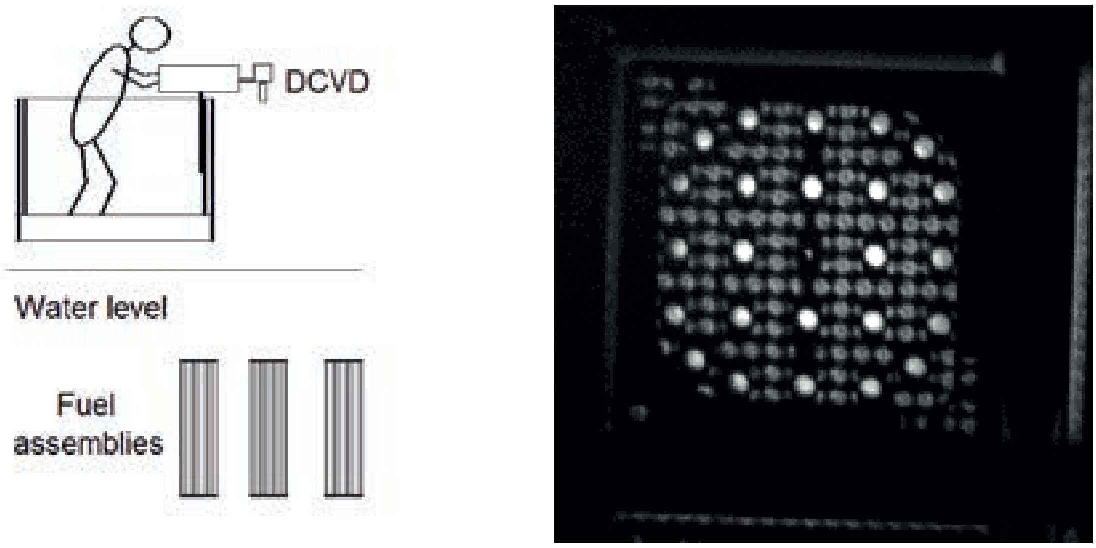


Figure 1 Left: the typical measurement situation when measuring irradiated nuclear fuel assemblies with a DCVD. **Right**: an example of a gray-scale DCVD image of a PWR fuel assembly.

been estimated from the fuel burnups and cooling times. Conclusions on whether the fuels are intact or suffer from partial defects are drawn after measurements have been performed. Currently, an inspector enters information about the fuels to be measured into a program that estimates the Cherenkov light intensity by interpolating precomputed data for fuels with varying burnup and cooling time. Entering the fuel data into the program doing the interpolation is often done manually, but can be done automatically if there are scripts available to read the data in the format provided by the operator.

As can be understood, the prediction method for the Cherenkov light intensity needs to be both fast and accurate in order to be useable in the field. The CPM, which takes only the final burnup and the cooling time since end-of-irradiation into account, has performed well in most measurement campaigns. However, it has not been able to accurately predict the Cherenkov light intensity from short-cooled fuel, with a cooling time shorter than a few years. For such short-cooled fuels, the irradiation history influences the Cherenkov light intensity to a relatively large extent. Furthermore, implementing a more detailed method will improve the capability of the DCVD to detect partial defects, by putting more stringent limitations on both the expected intensity value and its uncertainties. For these reasons, the method presented here was developed.

1.2 Next-generation prediction tools

To model the Cherenkov light generation in a fuel assembly, a Geant4 [2] based simulation toolkit has previously been developed [3]. This toolkit has now been updated to work with the latest version of Geant4, and scripts have been developed to launch the simulations and collect the data with limited efforts from the user of the toolkit. This work is a continuation of the work in [4], further developing the models and procedures used for the simulations of predicted Cherenkov-light intensities.

The updated toolkit uses a Geant4 standard physics list, with optical photon physics added separately. The gamma source in the simulations can be chosen to be one of the following: 1) a monoenergetic source, 2) an arbitrary spectrum provided in a format the program can read, or 3) an output file from the fuel depletion calculation program ORIGEN [5]. The geometry of the fuel assembly, including rods with cladding and a fuel box surrounding the fuel is specified in an input file. The standard simulation settings are applicable to most cases, but via input files the user may alter settings such as cut-off energies of gammas and electrons, and what data to save during a simulation run.

2. Current prediction method (CPM) for predicting the Cherenkov light intensity

The CPM used to predict the Cherenkov light intensity from an irradiated nuclear fuel assembly is based on a method developed by Rolandson [6]. In this method, the Cherenkov light intensities for a selection of boiling water reactor (BWR) fuels with varying burnups and cooling times were obtained through Geant3 simulations. In the analysis, these results are used to interpolate the expected Cherenkov light emission intensity for a fuel assembly, given its burnup and cooling time. The method was later extended by others to work for a larger range of burnups and cooling times, as well as to give predictions for short-cooled fuel, with cooling time shorter than one year.

2.1 Description of the CPM

The simulations of the data that form the basis of the CPM were performed in two steps. The first step was to calculate the concentration of fission product isotopes in a fuel assembly using a fuel burnup calculation program, and create a gamma spectrum as emitted by the fuel. The second step was to transport the gamma rays from their place of emission inside the irradiated nuclear fuel, which includes tracking them through their interaction with e.g. electrons in water and in the Cherenkov light generation process in the surrounding water. The last step also included the transport of the Cherenkov photons to the place of detection.

The first step was done using the ORIGEN fuel depletion code, and as a result the gamma spectra emitted from the selected BWR assemblies with different burnups were calculated. In the original studies the fuels were irradiated during four to six irradiation cycles, with each cycle having an irradiation period of 330 days followed by 35 days of cooling. The length of the final cycle was adjusted so that the fuels had the desired total burnup. The power level was chosen to be equal at all times of irradiation. However, for high-burnup fuels, more irradiation cycles were added, while for low-burnup fuels, four irradiations cycles were used with a lower power level. After irradiation, the gamma spectrum was saved for several cooling times in the range of 1 to 50 years. To simplify the calculations, the original investigations concerned only the contributions from the six isotopes Y-90, Rh-106, Cs-134, Cs-137, Pr-144 and Eu-154, which together contributed to more than 96% of the total gamma ray intensity at energies which may result in Cherenkov light being produced for the burnups and cooling times under consideration. All these nuclides are gamma emitters, except for Y-90. This isotope emits high-energy electrons that cause Bremsstrahlung in the fuel, which contributes to the total gamma spectrum. All recent updates of these simulations, including results presented here, include not only these six isotopes, but the full inventory of gamma emitters. This is especially important when extending the simulations to more short-cooled fuel, where many short-lived isotopes are present and contribute to the total Cherenkov light intensity.

The second step was to simulate the transport and interaction of the gamma rays in the fuel geometry. This was done in the Monte Carlo code Geant3. The code simulated the gamma ray interactions with surrounding matter (fuel, cladding and the water), the creation of electrons and the generation of Cherenkov light. The propagation of the Cherenkov photons to a detector position 5 m above the fuel was also simulated. A "shadow factor" was also introduced, to take into account the effect of spacers and top structures in the fuel. This factor was multiplied with the simulated intensity to get an estimate of the measured intensity. Later simulations used a simplified geometry, and the total emitted Cherenkov light intensity was used as an estimate rather than a simulated intensity at a detector position.

2.2 Limitations of the CPM

The CPM has worked well so far. However, since it does not take into account the irradiation history, it has proven to be less accurate in predicting the Cherenkov light intensities from short-cooled fuel. Furthermore, a simplified geometry is used and the results from the simulations of a specific BWR fuel type are taken to approximately describe all types of fuels. The impact of these two factors is described in the following two sections.

2.2.1 Irradiation history

The CPM works very well when the decays of Cs-137 are the dominating contribution to the emitted Cherenkov light. Since Cs-137 is long-lived with a half-life of 30.2 years and since it is proportional to burnup, the knowledge of the burnup and cooling time of a set of fuel assemblies is sufficient to estimate their relative Cherenkov light intensities. However, for fuels with a cooling time on the order of or less than two years, the irradiation history of each fuel assembly and its location inside the reactor core becomes more important. A typical fuel in a power producing reactor is placed near the center of the reactor core for the first cycles, where it has a relatively high power level, and spends the last cycles near the edge of the core, with a lower power level. For short-cooled fuels, the short-lived gamma-emitting isotopes which were generated in the last irradiation cycle are still present. This means that the assumption of an equal power level in all irradiation cycles is not fully valid, especially if the burnup of the last cycle deviated significantly from the average of the previous ones.

While the "typical" fuel has a high burnup in the first cycles and low burnup in the last ones, it is also common that fuels may have an irradiation history which differs from this. If a fuel spends a cycle outside the reactor before being irradiated again, or if the final cycle is high-power, this can greatly affect the gamma spectrum of the fuel at discharge. Thus, for short-cooled fuel, the fuel irradiation history must be taken into account to accurately predict the Cherenkov light intensity.

2.2.2 Geometry

The foundation for the CPM is based on simulations of 8x8 BWR fuel. It is currently being investigated to what extent, and with which accuracy, the results can be applied to predict the Cherenkov light intensity from other types of irradiated nuclear fuels. In addition, it is being investigated whether the simulations can be simplified by simulating and extrapolating the Cherenkov light emitted by one single fuel rod, rather than a full assembly, in order to speed up the process. Work is ongoing in both areas, but it is worth mentioning that by using simplified geometries there is a risk of neglecting differences between different fuel types. This may e.g. impact the Cherenkov light generation process and the transport of the Cherenkov photons from their place of emission to the DCVD, and hence increase the errors in the predicted intensity for other fuel types. Thus, there is a need for a prediction method which also takes the fuel geometry into account.

3. Proposed next generation method (NGM) for predicting the Cherenkov light intensity

In this paper, it is argued that more accurate predictions of the Cherenkov light emission from a fuel assembly can be obtained through simulation of its actual fuel irradiation history and detailed Monte Carlo modelling of the Cherenkov light generation in the entire fuel assembly, and for this reason the NGM has been developed. As in the case of the CPM, the fuel depletion step is done in ORIGEN, and the particle transport is done in Geant4. The modelling of the particle transport takes into account the full fuel geometry including all rods, the cladding and the fuel box. The process of repeating the generation of the source term (i.e. the gamma spectrum) for different fuel geometries also allows for an investigation of possible differences between different fuel types. To speed up the prediction process, the Cherenkov light production due to gamma rays of a given energy in an assembly can be pre-computed, and given a gamma spectrum from e.g. ORIGEN, the Cherenkov light intensity can be estimated quickly based on the pre-computed values.

3.1 Simulating Cherenkov light from BWR and PWR fuel geometries

To investigate the difference between the CPM and the proposed NGM, and to study if the gamma spectrum and Cherenkov light production depends on fuel type, simulations have here been performed for an 8x8 BWR and a 17x17 PWR fuel design. The fuel history simulated in ORIGEN was chosen to be rather similar to the one used in [6] so that the results may be compared. However in this work all the cycles were of equal length, while the previous work adapted the length of the final cycle in order to meet the desired total burnup.

For all simulations, an initial enrichment of 2% was assumed, to allow comparison with the earlier results. Fuels with 10, 20 and 30 MWd/kgU burnup were irradiated for four cycles, where each cycle had 312.5 days of irradiation and 46 days of cooling. The power levels for the three burnups were 8, 16 and 24 kW/kgU, respectively. For the 40 MWd/kgU case, the power level remained at 24 kW/kgU, and the fuel was irradiated for 5 cycles. For all burnup levels, separate gamma spectra were saved at 0.25, 0.5, 1, 2, 3, 5, 7, 10, 15, 20, 30, 40, 50 and 60 years of cooling time after discharge. The same irradiation histories, initial enrichment and power levels were used for both the BWR and the PWR studies.

The gammas rays from the fission products inside the fuel were generated in the vertical center of the full fuel rod length, and at randomly distributed positions in the horizontal fuel rod plane. The momentum directions of the gamma rays were isotropic. To save computer time, gamma rays with energy below 300 keV were not simulated, since simulations show that for gammas just below 300 keV only one or two Cherenkov photons are generated per 10 million gammas, which is negligible. Further, electrons with energy less than 257 keV were discarded in the simulations, since these have too low energy to produce Cherenkov light. Once a gamma ray had energy lower than 257 keV it was also discarded, since it cannot produce electrons with sufficient energy to produce Cherenkov light.

Geometrically, the 8x8 BWR fuel geometry was matched to the one used in [6], with the same fuel and cladding diameters and rod pitch (center distance between rods). The chosen PWR geometry was a 17x17 Westinghouse type with water filled guide tubes for control rods and a central instrumentation tube. The geometrical fuel information for the BWR and PWR fuels is given in Table 1. The inner radius of the cladding is chosen to be the same as the fuel pellet radius, corresponding to a closed gap in between the fuel and the cladding. Since the fuel types simulated are rotationally symmetric, it was sufficient to simulate one octant of the fuel, and the information could be used to predict the Cherenkov light contribution from the rods that were not simulated.

Property	BWR 8x8	PWR 17x17
Fuel size [mm]	130 * 130 * 3985	214*214*3852
Pellet radius [mm]	5.22	4.09
Cladding outer radius [mm]	6.13	4.75
Pitch [mm]	16.3	12.6

Table 1: Geometry details of the implemented BWR and PWR geometries.

The Monte Carlo simulations were run on the UPPMAX computer cluster at Uppsala University, with each fuel rod submitted as a separate job, enabling all rods to be simulated in parallel. For each rod, 10 million fission product gamma rays were simulated, with the energy distribution given by the gamma spectrum from ORIGEN. The results of all the separate jobs were merged, to give a total emitted Cherenkov light intensity for the given gamma spectrum of the fuel. The statistical uncertainty in the total emitted Cherenkov light intensity of an assembly with a given gamma spectrum, due to the Monte Carlo nature of the simulation, was estimated to be less than 0.4% for all BWR simulations, and less than 0.1% for all PWR simulations.

3.2 Pre-computing the Cherenkov light intensity of a fuel assembly

While a simulation of a complete assembly is expected to give accurate results, such simulations are too comprehensive to be executed during a measurement campaign. One way to speed up the process of predicting the light intensity is by pre-computing the Cherenkov light intensity for each rod at a number of different gamma energies. These pre-computed intensities can then be combined with the assembly gamma spectrum to quickly obtain an estimate of the Cherenkov light intensity in the fuel.

The first step in predicting the Cherenkov light intensity is to use a program such as ORIGEN to simulate the fuel irradiation history, to obtain a gamma spectrum of the assembly. The gamma ray spectrum is then combined with pre-computed values of how much Cherenkov light is generated in an assembly by gammas of various energies. Since the gamma spectrum is typically binned, the pre-computed values can be the number of Cherenkov photons generated in a fuel assembly per gamma quantum in each bin which occurs in the gamma ray spectrum. This makes it very easy to estimate the Cherenkov light intensity from an assembly, since there is information about the gamma ray intensity per bin, as well as the Cherenkov light production per gamma for each energy bin.

To test this method and to compare it with the simulations done in section 3.1, simulations were run for both a BWR and for a PWR fuel assembly. The simulations were run for one octant of the fuel assembly, using the symmetry of the fuel to obtain the intensity values for the other rods. For each rod and for each gamma ray energy bin, a simulation of 10 million gamma rays was run. This corresponds to about 4000 CPU-hours to simulate both the BWR and the PWR fuels. This work is extensive, but only needs to be done once for every fuel geometry.

Due to the large amount of simulated particles, the statistical uncertainties in the resulting Cherenkov light emission are very low, typically around 0.03% per gamma ray energy bin. Although the statistic uncertainty in the gamma ray spectra from ORIGEN is low [7], larger systematic uncertainties arise due to not modelling e.g. the complete fuel history including every control rod movement during irradiation, and one may thus relax the statistical precision somewhat in the Cherenkov emission simulations without affecting the overall uncertainties significantly. Accordingly, simulating a complete assembly may be done in a few hundred CPU-hours while still having a statistical uncertainty much lower than the systematic uncertainty of the gamma ray spectrum.

4. Results

This section presents the results of the simulations. The first subsection compares the results of BWR simulations using the CPM in [6], with the complete assembly simulations done using the NGM here. In the next subsection, a comparison is made between the Cherenkov light intensity from BWR and PWR fuel assemblies with identical irradiation history.

4.1 Comparison of the current and next generation methods

The results of the CPM and the NGM are shown in Figure 2, where the results have been scaled to be equal at 10 years. As can be seen, the new simulations stretch into shorter cooling times than the results from the CPM.

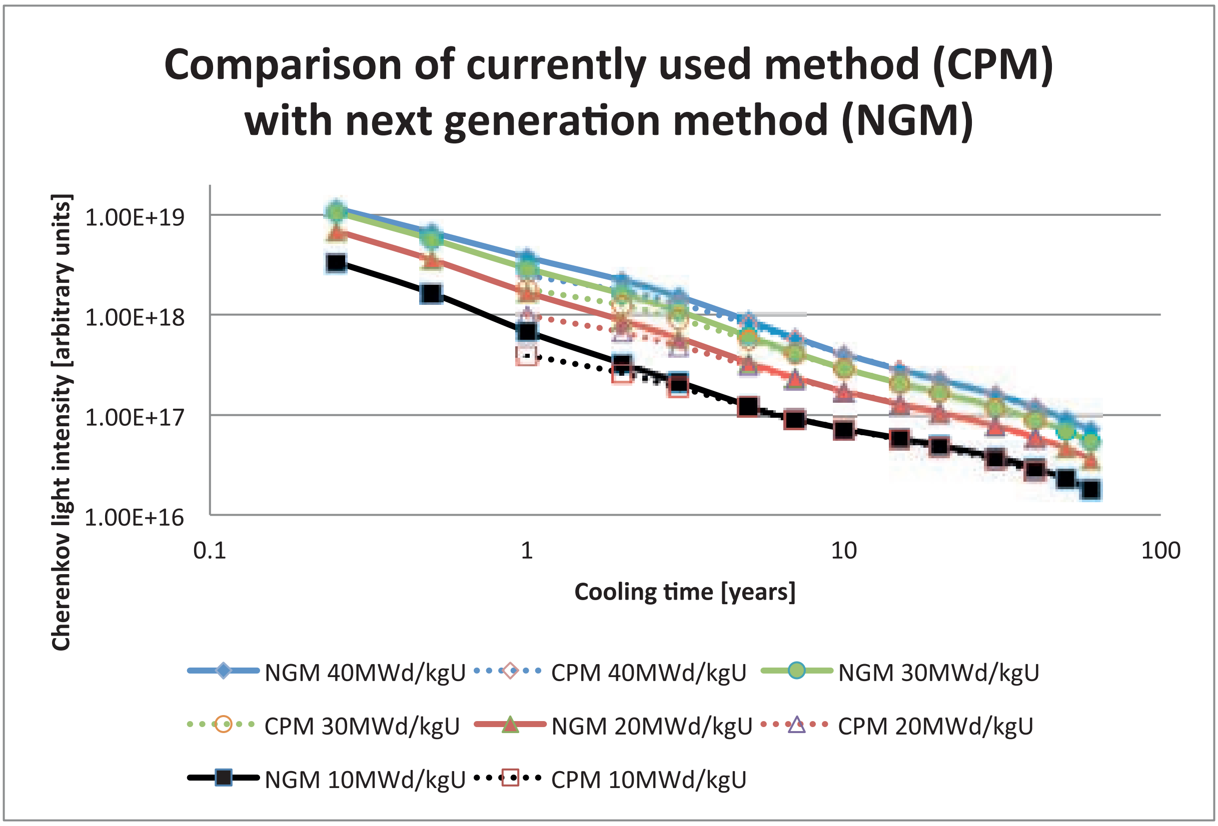


Figure 2: Comparison of a full BWR assembly simulation with the currently used method (CPM) [6] and the next generation method (NGM), normalized to 10 years cooling time. Statistical uncertainties of simulated values are smaller than 0.4% of the value.

The results from the CPM and the NGM are quite similar for fuel with a cooling time of more than 10 years, but for short-cooled fuels the results deviate. The simulations done here suggest a higher relative intensity compared to the CPM. This can be expected since the new simulations include the contribution of all isotopes in the fuel, while the older results only considered six long-lived isotopes. The difference may also depend on small differences in the fuel irradiation history used. Investigating how the irradiation history affects the Cherenkov light intensity is the subject of future work. Further, the differences may also be due to updates in nuclear cross sections in ORIGEN and updates in physics models used in Geant4 compared to the older versions of the code.

The results also differ from those found in [4], where a somewhat lower intensity was found. The cause of this is under investigation, but possible reasons are improved methods to introduce ORIGEN spectra into Geant4 and updates in the physics models used in Geant4.

Done in an automated way, the ORIGEN burnup calculations and the estimation of the Cherenkov light intensity using pre-computed simulations can be performed in a few seconds per fuel, which is fast enough to be practically useable during measurements.

4.2 Cherenkov light emission from different fuel types

The CPM is based on simulations of a BWR fuel respectively on simulations of a single rod. If the results of such simulations are applied to other types of fuels, the predictions become more uncertain since the effect of fuel geometry on Cherenkov light production is not taken into account. To investigate what effect the fuel geometry has on the Cherenkov light production, simulations were run for a BWR and a PWR fuel assembly with identical irradiation history.

Comparing the simulated emission of Cherenkov light from a PWR fuel to that of a BWR fuel reveals that the intensity profiles as a function of time differ, as shown in Figure 3. With a normalization of data to 10 years' cooling time and the same uranium mass, the Cherenkov light emission appears to be higher for PWR fuels as compared to BWR fuels for short cooling times. This is most noticeable for the simulated fuel assemblies with low burnup, where the difference is largest at 2-8 years, depending on the burnup. For very short cooling times of less than one year, the intensity is instead lower for PWR fuels, which may be explained by the difference in how short-lived isotopes are built up in BWR and PWR reactors. For a cooling time longer than 10 years, the Cherenkov light intensity for the PWR fuel appears to be lower compared to a BWR fuel.

These effects are on one hand due to differences related to the reactor core design of BWR and PWR reactors, such as the presence of void in the BWR reactor and on the other hand due to differences in the fuel geometry such as fuel and cladding size, pitch, and guide tubes which affect the amount of water inside the fuel assembly. The statistical uncertainty of the total Cherenkov light intensity in the BWR simulations were typically between 0.2% and 0.4%, while for the PWR case it was smaller than 0.1%, which means that the difference between the two are significant.

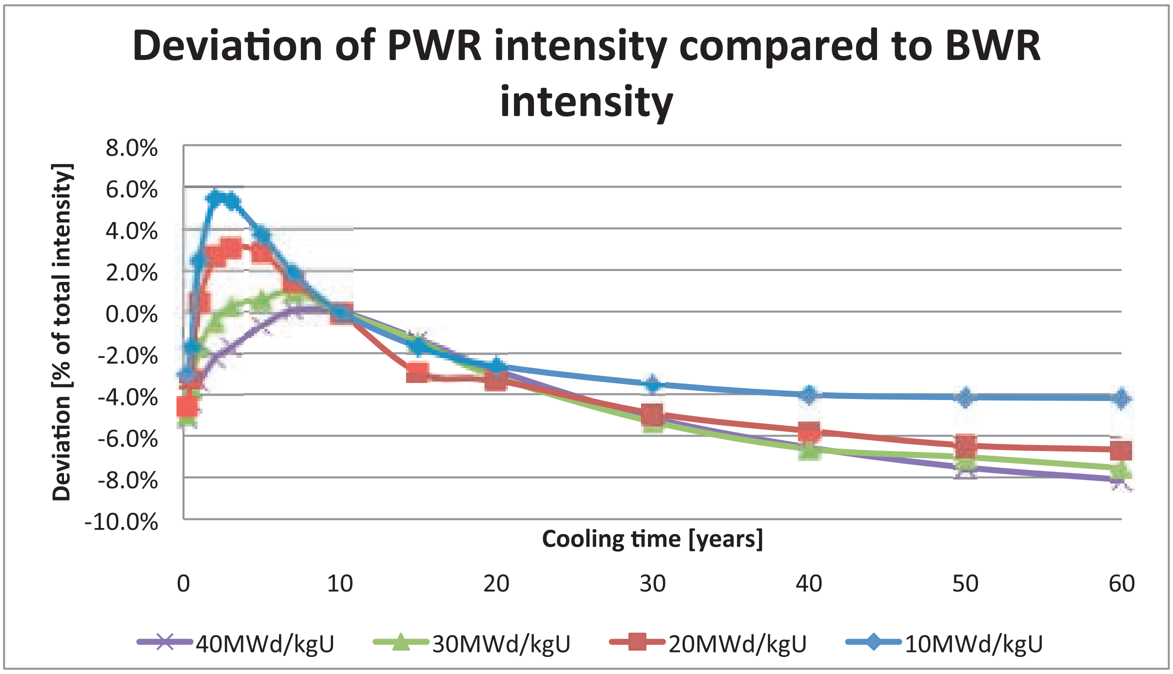


Figure 3: Deviation of the simulated emitted PWR Cherenkov light intensity compared to the BWR intensity, normalized to 10 years cooling time and the same uranium mass content. The uncertainties of the values are smaller than 0.5 percent units.

As a consequence, if Cherenkov light emission intensities based on BWR simulations are used to predict corresponding intensities for PWR assemblies or vice versa, a noticeable error is introduced. For accurate estimations of the Cherenkov light intensity, one may conclude that each fuel type should have its own set of intensity estimates.

5. Conclusions

This paper describes the currently used method for predicting the Cherenkov light intensity emitted from irradiated nuclear fuel assemblies in wet storage. It also presents results from the new proposed method for predicting the Cherenkov light intensity generated in an assembly for both BWR and PWR fuels, and suggests a new, quick and accurate way to obtain the same information. The new method makes use of pre-calculations to estimate the contribution to the total Cherenkov light intensity in the assembly for gamma-rays of various energies from each rod. It takes into account both the fuel history and fuel geometry, while still being fast enough to allow for use during field measurements or when only limited computational resources or time is available. The increase in prediction accuracy can aid in setting more stringent limits on how much a measured intensity may deviate from a predicted value, which, in turn, may be used to improve the partial defect detection capability of the DCVD. The simulations performed also show that the new method gives very similar results to the currently used method for fuels with a cooling time longer than 10 years, but shows a different behavior for short-cooled fuel.

The new method requires computationally expensive pre-calculations, done separately once for each fuel type, preferably on a computer cluster. For long-cooled fuel, or for fuel where a detailed irradiation history is unavailable, a standard irradiation history may be applied. Using such a standard fuel history, the intensity estimates can be obtained with the same amount of work as the currently used prediction method. However, for short-cooled fuels, with a cooling time on the order of one or two years, the irradiation history must be taken into account, which is possible using the methodology suggested here. It is also possible to automate the input of the fuel history into ORIGEN, running it, and extracting the resulting gamma ray spectra. A modern laptop is capable of producing around 2000 such estimates per hour, which includes the ORIGEN simulations of the fuel history. This will allow for predictions taking into account the fuel irradiation history, while still not adding to the workload of converting the operator declaration to a Cherenkov light intensity estimate.

6. Outlook

One possible extension to the suggested method is to not only pre-calculate the total emitted Cherenkov light intensity, but to also handle the transport of the Cherenkov light to a detector position. This will further increase the accuracy of the method, since it predicts what the DCVD can actually detect. This will however require additional information on the geometrical details of the fuel assembly such as spacers and top structure to be taken into account, as well as the axial burnup distribution and the absorption of light in water. A related question is to what extent this information is available to an inspector on site. Furthermore, material deposits on fuel rod surfaces (CRUD) may alter the optical properties and thus change the absorption and reflectivity of UV light, which must also be taken into account. Also, if the prediction takes into account the light intensity which the DCVD can measure, it may also be possible to directly compare the Cherenkov light intensity of fuel assemblies of different types.

7. Acknowledgements

This work was supported by the Swedish Radiation Safety Authority (SSM), under contract SSM2012-2750. The computations were performed on resources provided by SNIC through Uppsala Multidisciplinary Center for Advanced Computational Science (UPPMAX) under project p2007011.

8. References

[1] International Atomic Energy Agency (IAEA), "*Safeguards Techniques and Equipment: 2011 Edition,*" IAEA/NVS/1/2011, 2011.

[2] S. Agnostelli et. al., (the Geant4 collaboration), "*Geant4 - a simulation toolkit,*" Nuclear Inst. and Meth. in Physics Research Section A: Accelerators, Spectrometers, Detectors and Associated Equipment, Volume 506, Issue 3, 1 July 2003, Pages 250-303.

[3] S. Grape, S. Jacobsson Svärd, B. Lindberg, "*Verifying nuclear fuel assemblies in wet storage on a partial defect level: A software simulaiton tool for evaluating the capabilities of the Digital Cherenkov Viewing Device,*" Nuclear inst. and Meth. A, Volume 698, 11 January 2013, Pages 66-71, ISSN 0168-9002, 10.1016/j.nima.2012.09.048.

[4] S. Grape, S. Jacobsson Svärd, "*Recent modelling studies for analysing the partial-defect detection capability of the Digital Cherenkov Viewing Device,*" ESARDA bulletin no 51, December 2014.

[5] S. M. Bowman et. al., "*ORIGEN-ARP, a fast and easy to use source term generation tool,*" Paper submitted to ICRS - 9, Ninth International Conference on Radiation Shielding, August 6, 1999.

[6] S. Rolandson, "*Determination of Cherenkov light intensities from irradiated BWR fuel,*" IAEA task ID JNTA0704, SKI Report #: SE 1-94, 1994.

[7] J.J. Klingensmith, I.C. Gauld, "*ORIGEN-S Gamma Decay Spectra Characterization and Benchmarking,*" Trans. Am. Nucl. soc. 94, 385-387, 2006.

Experimental Assessment of a ^{6}LiF:ZnS(Ag) Prototype Neutron Coincidence Counter for Safeguards

H. Tagziria[a1], M. Foster[b], M. Schear[c], D. Ramsden[b], G. Dermody[b], B. Pedersen[a], and P. Schwalbach[d]

[a]European Commission, Joint Research Center, ITU-Nuclear Security Unit, I-21027 Ispra, Italy
[b]Symetrica Security Ltd., Roman House, 39 Botley Road, Southampton, SO52 9AB, UK
[c]Symetrica Inc., 63 Great Road, Maynard MA 01754
[d]European Commission, DG-ENER, Nuclear Safeguards - Unit E1, Section Measurement Systems, On Site Laboratories, Cooperation Programmes, EUFO 3477, L-2557 Luxemburg

[1]**Contact**: hamid.tagziria@jrc.ec.europa.eu

Abstract

A prototype 3He-free neutron coincidence counter for safeguards applications has been developed and built following comprehensive Monte Carlo modeling. It consists of eight compact 6LiF:ZnS(Ag) thermal neutron absorbers (or blades) dispersed in four moderating slabs surrounding the sample chamber.

This paper describes the results of an extensive campaign of measurements carried out at the JRC in Ispra to validate the Monte Carlo models, characterize and calibrate the counter in order to assess the suitability of the technology as an alternative to ^{3}He based ones. Its compliance with safeguards requirements regarding a number of important parameters such as neutron efficiency, die-away time, gamma rejection, dead-time amongst others is also evaluated.

The counter successfully took part in an inter-comparison measurement campaign at the JRC in Ispra (Italy) within the International Safeguards Workshop on ^{3}He alternatives in October 2014 attended by a number of laboratories and research institutes from Europe, IAEA and USA with Japan as observers.

The performance of the counter is compared to that of a commonly deployed HLNCC-II counter which makes use of now scarcely available ^{3}He gas.

Keywords: NDA; Nuclear Safeguards; Neutron Coincidence Counting

1. Introduction

Over the past few decades, non-destructive assay (NDA) in the field of nuclear safeguards has relied on the neutron coincidence counter (NCC), using the time-correlation of fission neutrons to produce measurements of fission rate from which plutonium mass can be calculated. Such systems consist of a sample cavity surrounded by He^3 proportional counters embedded in high density polyethylene (HDPE), with attached electronics to carry out the time-delayed coincidence measurement. The “classic” design is represented by the HLNCC-II in service around the world [1].

However, in recent years the supply of ^{3}He has been outstripped by demand [2], leading to a perceived shortage in many fields, including nuclear safeguards. To mitigate this problem extensive research has been carried out worldwide into alternatives, focussing mostly on ^{10}B and ^{6}Li thermal neutron detectors or fast neutron detection using organic scintillators.

This paper describes research by JRC and Symetrica into a ^{6}Li-based system, utilising the safeguards and NDA expertise of JRC and the detector development expertise of Symetrica. The latter has demonstrated excellent performance in terms of gamma-ray rejection and sensitivity when applying ^{6}Li-loaded scintillators in systems ranging from roadside portals to handheld neutron detectors [3]. It was then a natural step to apply Symetrica compact thermal neutron detectors to the neutron coincidence counting problem.

It was decided that the development effort would be staged. The first stage consisted of two parts: designing an NCC capable of outperforming the HLNCC-II using simulations in MCNPx; and developing and testing the thermal neutron detectors that would populate it. Since the thermal neutron detectors are a new design, this stage only involved making only a small number sufficient for testing the NCC in a partially populated mode. That testing was used to validate models and provide an improved estimate of the performance of the fully populated NCC.

In order to judge the success of this effort, a detailed characterisation and calibration campaign of the partly populated NCC was carried out at JRC, culminating in a comparison of plutonium mass estimates against a benchmark set by an HLNCC-II.

The planned second stage will consist of using the results presented here to further develop the thermal neutron detectors and to improve the design of the NCC. A larger set of thermal neutron detectors including improvements will be manufactured and used to test the full NCC, again against the HLNCC-II benchmark.

2. Thermal Neutron Detector Design

2.1 Neutron Detector "Blade" Description

The neutron coincidence counter under test relies on a number of thin thermal neutron detectors utilising ^{6}LiF/ZnS scintillators and silicon photomultiplier (SiPM) readout, dubbed "blades" for their form factor. Figure 1 shows images and the physical parameters of the thermal neutron detector "blades" developed for this project.

Each blade consists of a sensitive element and processing electronics. The sensitive element is made up of two ^{6}LiF/ZnS screens (EJ-426) sandwiching a wavelength shifting PVT plate of 500 x 60 x 3mm (EJ-280), both sourced from Eljen Technologies. The screens cover the whole face area of the wavelength shifter and have an average thickness of 0.75mm. Scintillation light from the screens is transmitted by the wavelength shifter to a row of silicon photomultipliers (S10931-050P) supplied by Hamamatsu Photonics optically bonded to one end.

The processing electronics is a single-board solution that includes the following functions:

- Preamplifier to provide current to voltage conversion and signal conditioning.
- Temperature stabilized SiPM bias supply in the range 66V to 76V using a linear function of 56mV/°C. A temperature sensor is placed near the SiPMs for this purpose.
- Neutron/gamma/SiPM-noise pulse shape discrimination (PSD). An internal 20MHz clock gives a 50ns time resolution.
- Simple internal fault detection and reporting via a "device ready line" that can be picked up by attached electronics.
- Communication over SPI to set parameters such as SiPM bias and neutron/gamma discrimination thresholds. Internal memory holds these settings following calibration.
- A TTL output signal to indicate detection of neutrons. The pulse is +5V and 150ns in duration.

The electronics are placed at the end of the sensitive element. An aluminium enclosure fits over the electronics and is electrically connected to a thin aluminum case over the sensitive element to provide EMI shielding. A layer of black heat-shrink plastic then shrouds the aluminum case to provide a protective layer.

Discrimination between neutrons, gamma-rays and SiPM noise is based on the measured length of pulses against a programmable threshold. A paralysable dead time of 10μs follows each neutron detection TTL. Each blade is self-contained and requires only a power line to operate. In return it provides a ready line to indicate that it is operating, and a TTL pulse for each neutron detected.

2.2 Aggregation of Blade Detectors

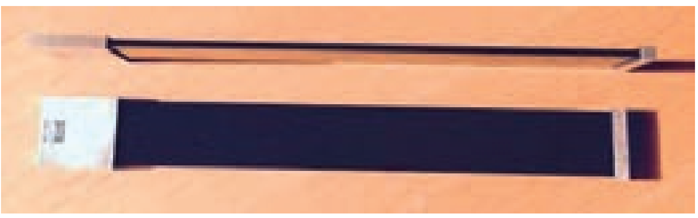
To facilitate testing of systems containing multiple blades, an Aggregator Unit was developed, and is shown in Figure 2. It serves to distribute power to up to 32 blades and contains a 32-input logical OR function with a single BNC output. This BNC can then be connected to counting circuits or pulse train analyzers for data collection. Blades are connected to the Aggregator Unit by 32 cables of 1.5m length that carry 12V power, TTL signals and ready lines.

2.3 Neutron Detector Testing

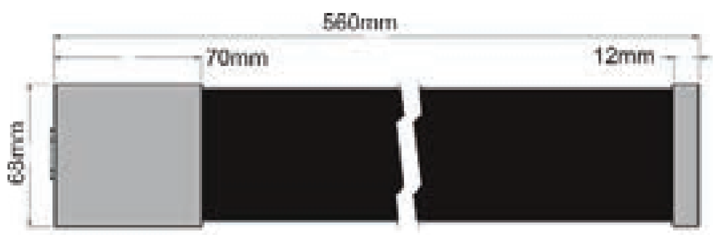
For this project a total of eight blades were assembled and tested for their thermal neutron detection efficiency and gamma-ray rejection. One of the set was also subjected to thermal tests to measure its variation in detection efficiency as a function of temperature.

2.3.1 *Thermal Neutron Detection Efficiency and Gamma-Ray Rejection*

The detectors used in the test module were calibrated to achieve the best thermal neutron detection efficiency whilst still meeting at gamma-ray rejection target limit of 10^{-7} or better, which means the probability of an incident gamma-ray being mistaken for a neutron was less than 10^{-7}.



- Mass = 430g
- Dimensions = 560 x 68 x 12 mm
- Power requirement = 9mA at 12V
- Temperature range = 10 to 30°C
- Neutron detection signal = +5V, 150ns TTL pulse



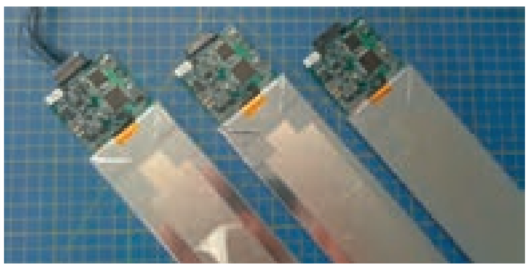


Figure 1: Clockwise from top left: An image of two blades, physical parameters of a blade, three blades showing their processing electronics, the dimension of a blade.

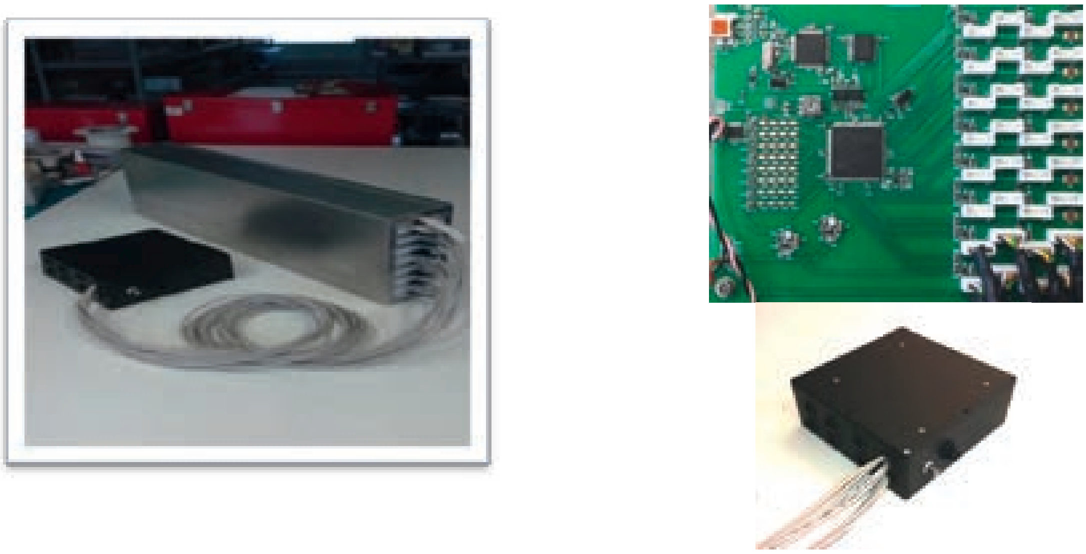


Figure 2: The Aggregator Unit (left) connected to eight blades in a moderator slab (right). Only eight cables have been fitted to the Aggregator for convenience.

The efficiency of each blade relative to the rest of the set was determined by irradiation with a 5.21ng ^{252}Cf source moderated in a 5cm thick HDPE cylinder (10cm Ø x 10cm L), as shown in Figure 3. Measurements were taken indoors since environmental scatter was not relevant for relative measurements.

For gamma sensitivity measurements, a 17.7mCi ^{137}Cs source was illuminating the largest face of the detector at a dose rate of 100mSv/hr. Results are shown in Table 1. Since there exists a tradeoff between these two measurements depending on pulse shape discrimination thresholds, the two measurements were carried out simultaneously.

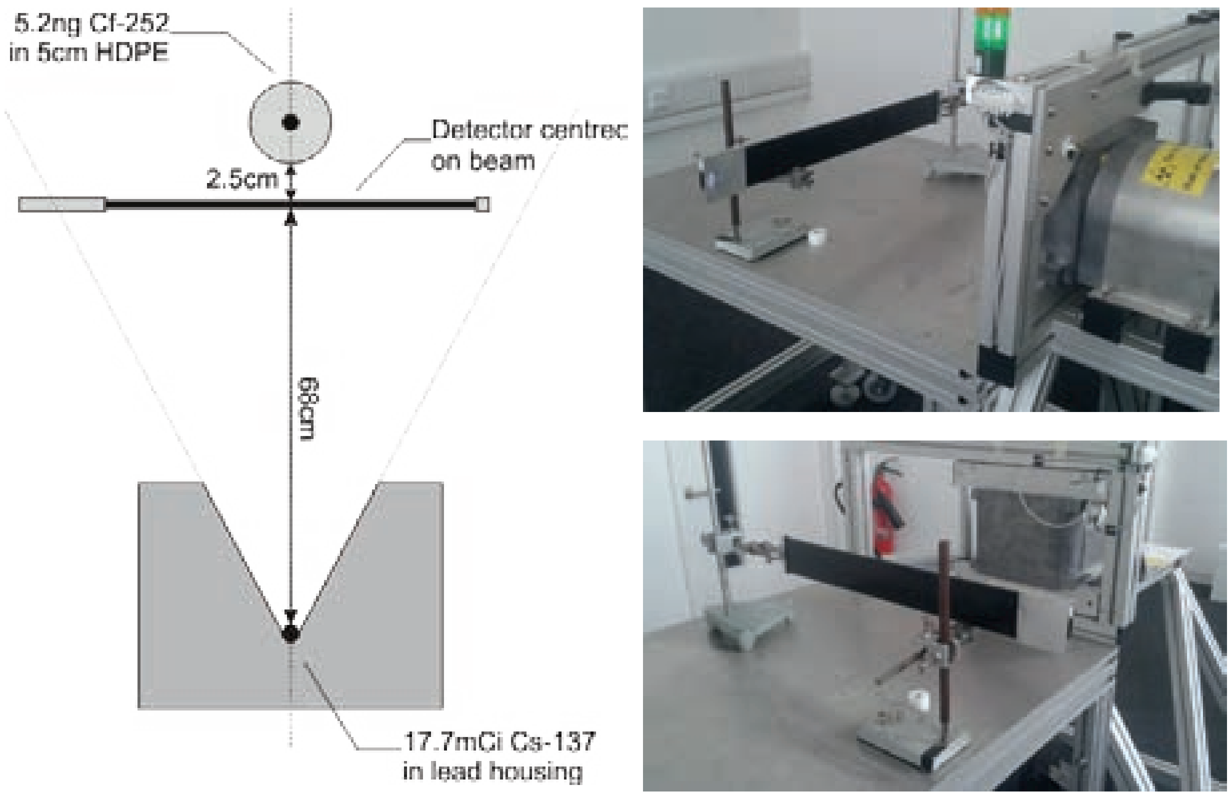


Figure 3: Left: a sketch of the detector in front of the gamma-ray source, with the neutron source in place, and the detector sensitive element centred on the gamma beam. Right: photographs of a blade in the test position.

Detector Serial Number	Relative Sensitivity (n/s)	Relative Sensitivity (% of average)	Gamma-ray Rejection (x 10^{-7})
140001	45.01 ± 0.39	101.3%	0.26 ± 0.11
140002	46.57 ± 0.40	104.8%	0.55 ± 0.12
140003	44.29 ± 0.39	99.7%	0.70 ± 0.14
140004	44.63 ± 0.39	100.5%	0.18 ± 0.10
140005	45.18 ± 0.39	101.7%	0.47 ± 0.11
140006	45.30 ± 0.39	102.0%	0.60 ± 0.13
140007	41.06 ± 0.37	92.4%	0.82 ± 0.14
140008	43.30 ±0.40	97.5%	0.50 ± 0.10

Table 1: Summarised results from the individual blade tests for relative thermal neutron detection efficiency and gamma-ray rejection.

The distribution of sensitivity is very tight, showing a standard deviation of only 3.7%. During this work it was found that the relative efficiency correlated strongly with the ^{6}LiF/ZnS screen thickness indicating that this sets the limit on uniformity, since each blade was tuned for the best sensitivity. If this distribution is later found to be too broad, then the highest performing blades can then be de-tuned slightly to narrow it.

2.3.2 Temperature Tests

In this test, blade S/N 140002 was placed in an environmental chamber and cycled in the range 10°C to 30°C multiple times. A neutron source in a 5cm HDPE moderator was also placed to provide a high count rate. During the cycle, the count rate was monitored. Figure 4 shows the results indicating a high degree of consistency over the range with no apparent hysteresis between cycles.

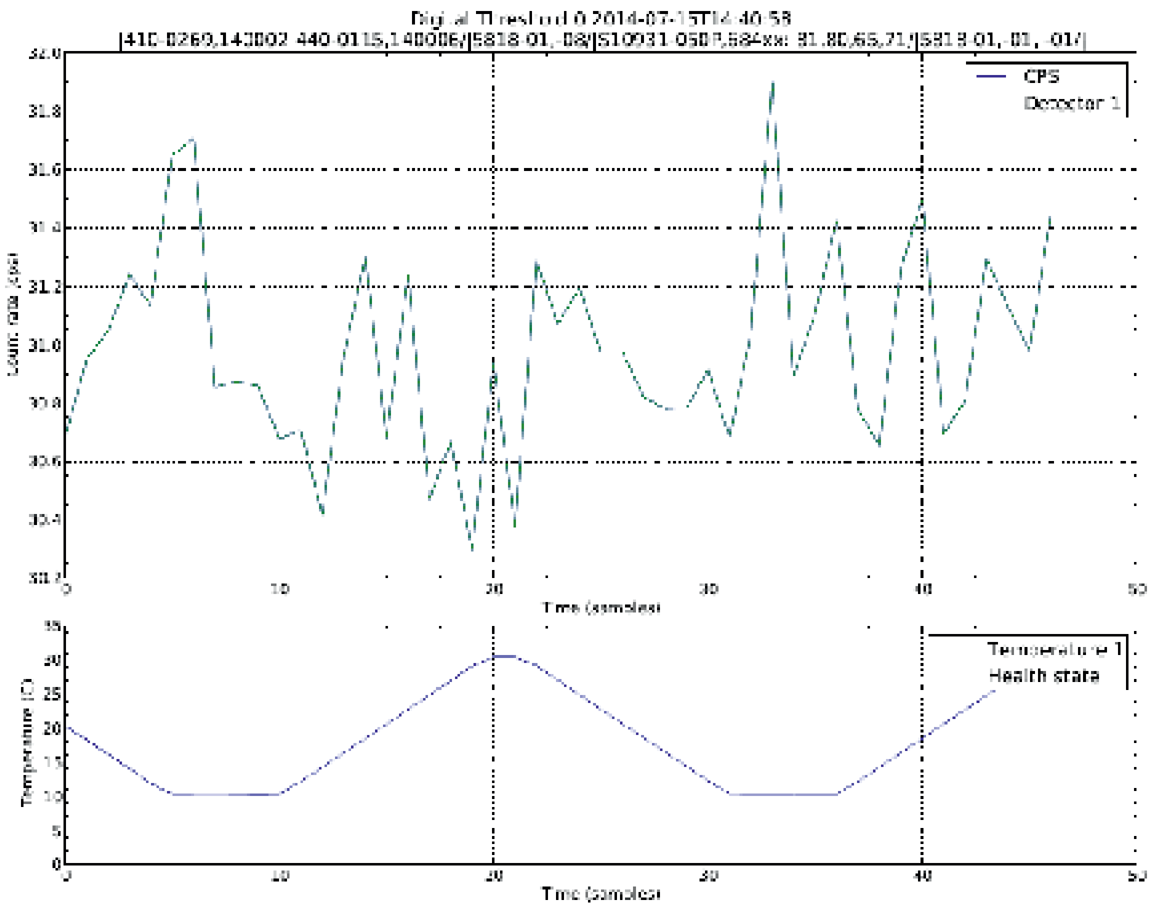


Figure 4: Data showing the consistency of neutron sensitivity as a function of temperature in the range 10°C to 30°C.

2.3.3 Readout Efficiency Measurement

The blades rely on a wavelength shifting process and pulse-shape discrimination to detect the capture of thermal neutrons in the ^{6}LiF/ZnS screens. These processes are not simulated in Monte Carlo models and so must be accounted for as the "readout efficiency" of the thermal neutron detectors to prevent the model overestimating the absolute efficiency of the NCC. Readout efficiency represents the probability of a neutron captured in the ^{6}Li producing sufficient signal to pass the discrimination threshold and result in a TTL pulse being counted.

The simplest way to estimate readout efficiency in the absence of a well-characterised thermal neutron flux is to place a detector in a low scatter environment and compare a measured absolute efficiency with a simulated one. That is,

$$E_R = \frac{\varepsilon_{abs,meas}}{\varepsilon_{abs,sim}}$$

where ε_{abs} is the absolute efficiency of the detector when exposed to a known neutron source.

This technique was used to measure the readout efficiency of the blades by placing all eight into a HDPE moderator that represented one quarter of the NCC (see Section 3). A sketch of the moderator is shown in Figure 5, along with a depiction of the model used. The blades were connected to the Aggregator Unit and the whole detector placed in an outdoor low-scatter environment. A ^{252}Cf neutron source of 5.21ng (± 0.9%) was placed at 8 cm from the centre of the largest face of the detector, as shown in Figure 5.

	Simulated	Measured
Absolute efficiency, ε_{abs}	6.07% ±.09%	5.17% ± 0.17%
Intrinsic efficiency, ε_{int}	21.4%±.09%	18.5%
Readout Efficiency, *RE*	**85.1%±2.9%**	

Table 2: Results of the readout efficiency measurement.

These results show a high readout efficiency of 85.1%, indicating a good optical path between the screens and SiPMs and that the pulse-shape discrimination algorithm performs well at finding neutron events amongst the SiPM noise.

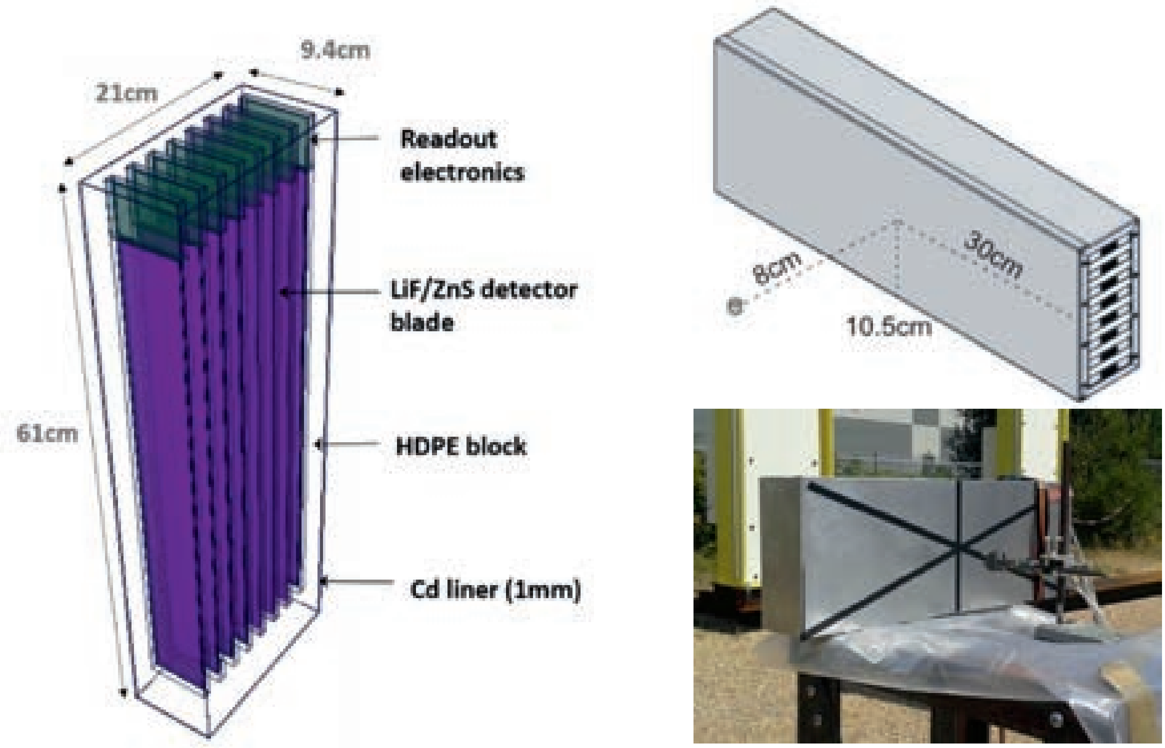


Figure 5: Left: A depiction of the modelled moderator with blades. Right: A sketch of the ^{252}Cf source placement, and a photograph of the measurement configuration.

3. Coincidence Counter Design

After having established the functionality of the blade detectors individually and within a slab module (see Figure 5), the next stage in the development consisted of designing and optimizing a full-scale coincidence counter build primarily consisting of four slab modules surrounding an assay chamber of square cross-section, as shown in Figure 6 (left). The modular design will also allow us to configure three modules as a collar for active measurements.

Monte Carlo N-Particle extended (MCNPX) simulations were performed to estimate the detection efficiency and the die-away time of the counter in order to compare to the HLNCC-II. Figure 6 (center) shows the MCNPX model of the counter showing the distribution of up to 32 blades in the HDPE moderating wall of the counter.

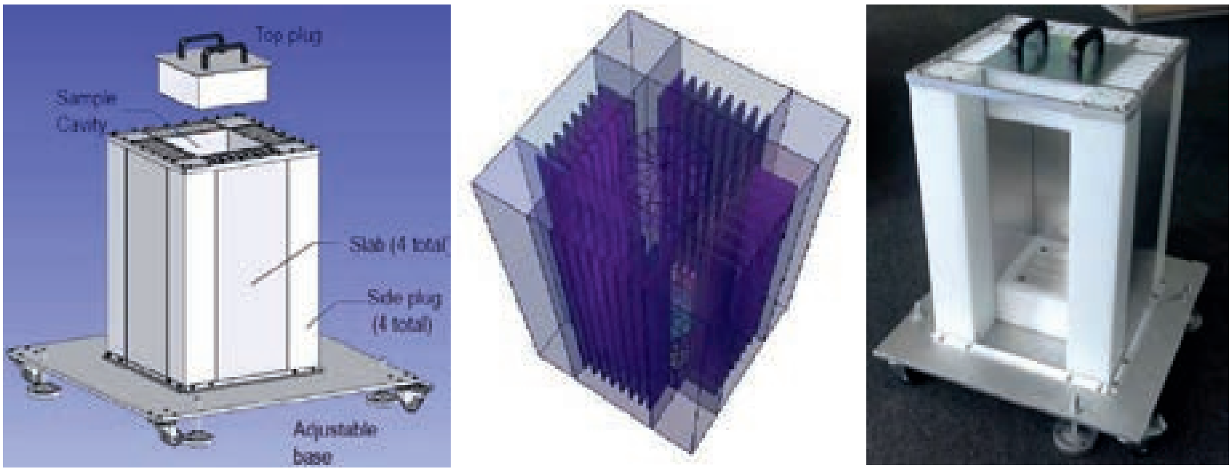


Figure 6: Full-scale NCC build illustration (L), MCNPX model (centre), and built with one slab removed (right)

Each slab module is enveloped in a cadmium sleeve to maintain a low die-away time by minimizing the thermal neutron albedo to the sample chamber. Top and bottom plugs were added comprising HDPE and aluminium neutron reflectors to give a good vertical profile. The design was completed by adding four corner posts of HDPE to increase absolute efficiency, without unduly increasing the die-away time since they are outside of the cadmium liners. The top and bottom plugs as well as the corner posts are visible in the right-hand panel of Figure 6.

The neutron source was simulated as a point isotropic ^{252}Cf in the center of the chamber. The simulated ^{6}Li capture probabilities were multiplied by the measured 85.1% readout efficiency. The die-away time of the counter was obtained using the built-in coincidence capture feature of MCNPX and the sequential gate width method, as well as fitting to the time distribution of the neutron population in the counter. A Figure-of-Merit, FoM, for comparison purposes was quantified as:

$$FoM = \frac{\varepsilon}{\sqrt{\tau}}$$

Where ε is the absolute efficiency and τ is the die-away time.

Figure 7 shows an iso-plot of FoM values for the blade counter design, normalized to the FoM of the HLNCC-II. The efficiency for a fully populated neutron blade coincidence counter (25.4%) is predicted to exceed that of the HLNCC (16.5%), and yet have a lower die-away time of 31µs, resulting in a coincidence counting figure-of-merit (FoM) of 4.56 compared to the 2.54 of the HLNCC-II.

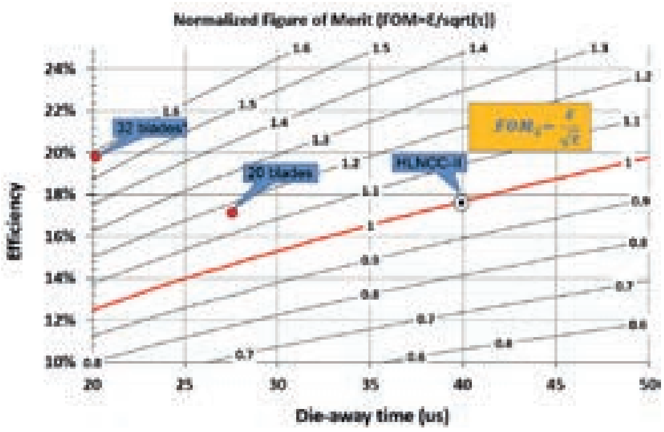


Figure 7: Simulated Figure-of-Merit for NCC compared to FOM of HNLCC-II

4. Experimental Verification

4.1 Counting and Analysis Electronics

During the measurement campaign, the blades were used with the Aggregator Unit which provided power and a logical OR of their output TTLs. As a consequence, the output of the system was a single BNC connector with output pulses of 150ns. The BNC output was connected to a PTR-32 from EK which was used to perform coincidence analysis. The PTR-32 provides the following functions using accompanying software:

- Recording of time interval data for replay and reanalysis
- Plotting of time interval histograms
- Calculation of Rossi-Alpha distributions
- Measurement of coincidence rates using the two-gate R and R+A method

Figure 8 shows a block diagram of the blades, Aggregator Unit and PTR-32.

4.2 Measurement of Basic Parameters

The NCC with eight blades was assembled in the PERLA laboratory at JRC in Ispra for testing. A number of HDPE "blanks" were manufactured that could be fitted into the spare blade slots to allow for reconfiguration of the NCC. Figure 9 shows the NCC in place with the Aggregator Unit connected to the blades. The blades were inserted with their electronics at the top of the NCC.

The basic performance parameters of the partially-populated NCC were measured, namely absolute efficiency and die-away time with ^{252}Cf. This was done with the blades and HDPE blanks in a number of different configurations. Results with these configurations are shown below in Figure 10.

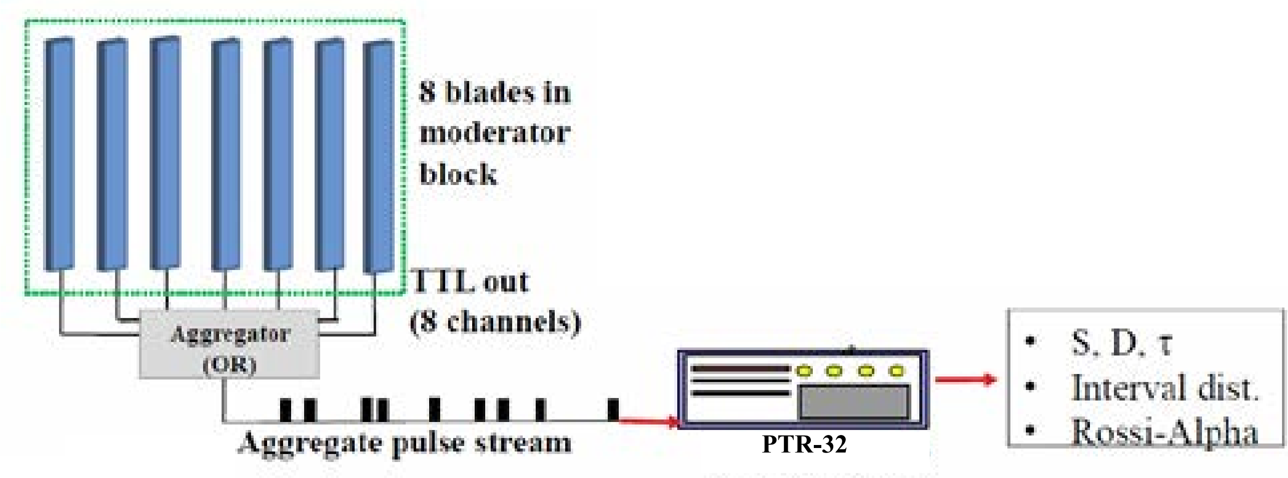


Figure 8: A block diagram showing the signal path used in this study.

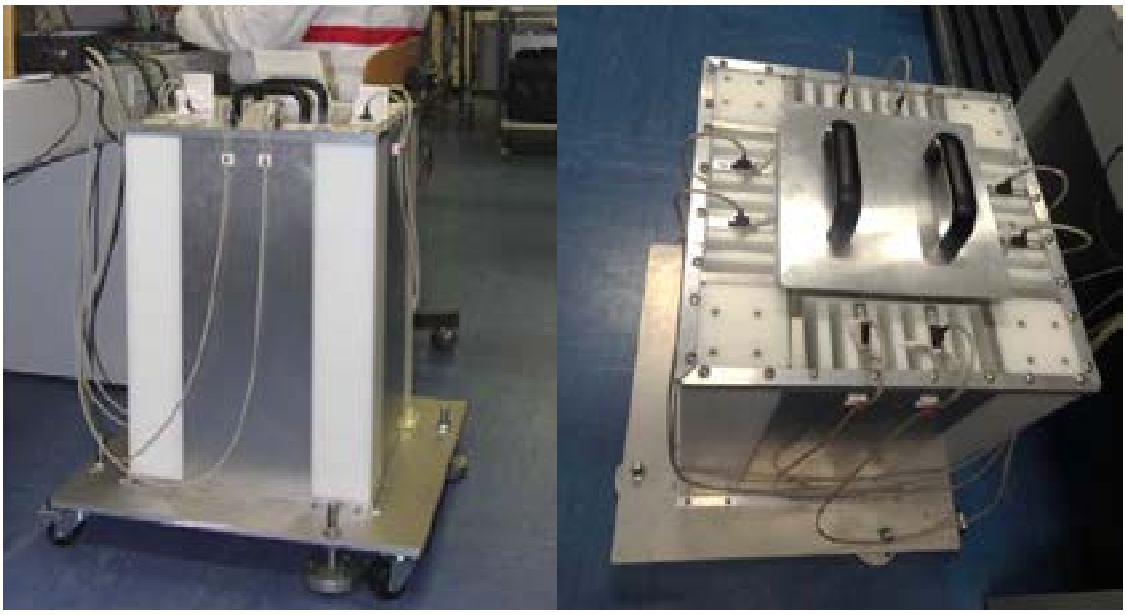


Figure 9: The NCC installed at JRC with eight blades and Aggregator Unit showing.

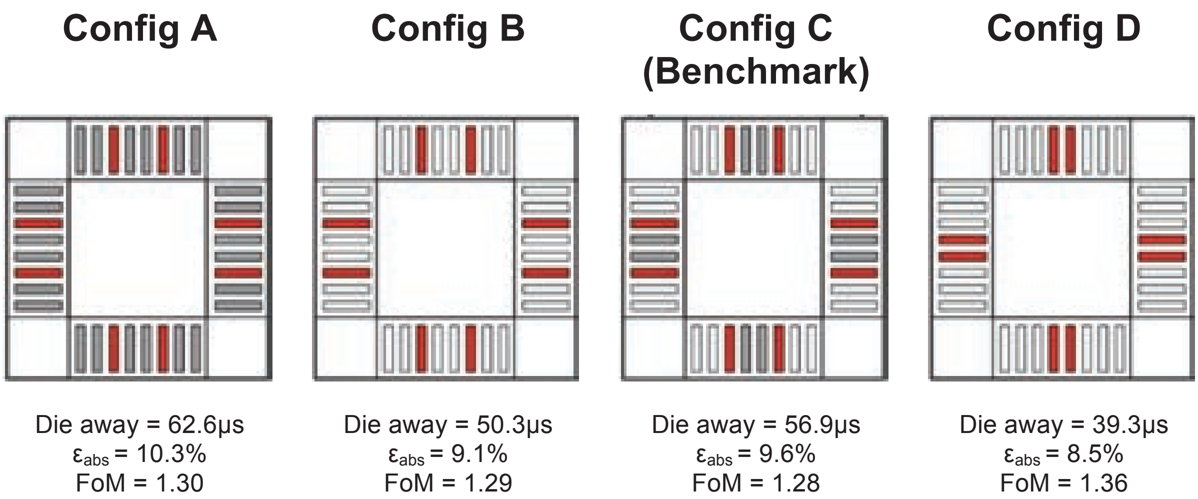


Figure 10: The basic parameters of the NCC as measured with the blades in various configurations. Absolute efficiency and die-away time were measured using ^{252}Cf. Blades are shown in red, blanks are shown in grey, and that empty slots are shown in white.

The efficiency achieved here is about half of what is expected of a fully-populated NCC, and varies only slightly with configuration. Clearly with a partially populated system, it is possible to trade off efficiency against die-away time using the HDPE blanks in the unpopulated slots. The presence of blanks has a greater effect on die-away time than on absolute efficiency and the best FoM is achieved by a configuration with the lowest die-away (Configuration D).

The efficiency of the NCC was measured for different neutron sources to understand the effect of neutron energy. Measurements were taken with blades in Configuration D.

Neutron Source and mean energy	Absolute Efficiency
Am-Li (alpha,n), 0.3 MeV	11.1%
Am-Be (alpha,n), 5.0 MeV	5.3%
Cf^{252} (fission), 1 MeV	8.5%
Pu^{240} (fission), 1MeV	9.9%

Table 3: Measured absolute efficiency for a range of neutron sources.

Am-Li clearly gives a very good efficiency due to its low average neutron energy. We can also see that Pu has a significantly higher absolute efficiency than ^{252}Cf. Am-Be has a very low absolute efficiency due to its high energy and the under-moderated nature of the NCC design, intended to keep die-away times to a minimum.

4.2.1 Comparison with Other He^3 Free Systems

The NCC was benchmarked alongside a number of other ^{3}He free systems in October 2014 at JRC [4,5]. Sample results as published are shown below.

The predicted FoM of the fully-populated NCC is very high, exceeding the HLNCC-II by ~80%, driven by a very high absolute efficiency. The results with the partially populated system are encouraging. However, FoM is a limited expression of the quality of a neutron coincidence counter, so further investigation is required to obtain a complete picture of how useful the NCC will be.

	HLNCC-II	PTI	Symetrica (8 blades)	Symetrica (32 blades) (simulated)	GE Reuter Stokes
Technology	^{3}He tubes	Numerous ^{10}B lined straws	^{6}Li loaded blades	^{6}Li loaded blades	Combined ^{10}B and ^{3}He proportional counter
Abs. Eff (%)	16.5	13.6	9.6	25.4	10.2
Die away time (s)	43.3	26	56.9	31.6	65.4
FoM	2.51	2.66	1.28	4.56	1.26

Table 4: Comparative results of three ^{3}He-free technologies and the HLNCC-II taken at JRC in October 2014

5. Characterization of Safeguards Relevant Parameters

5.1 Bias and Timing Characteristics

Bias is a key property of a neutron coincidence counter which consider rewording for clarity that suppress real coincidences amongst otherwise Poissonian neutron events. For this study, bias was measured for each blade individually and for the NCC as a whole. Am-Li sources were used for their Poissonian neutron output. The individual blade tests were conducted with all eight blades in the NCC with only one connected to the Aggregator Unit at a time. Results were analysed by taking Rossi-Alpha distributions of the recorded pulse trains using the PTR-32 software, and by calculating bias with the equation:

$$Bias\ (\%) = \frac{(R+A)-A}{A} \times 100.$$

Bias measurements were also taken at two count rates separated by an order of magnitude to assess variation with count rate.

5.1.1 Individual Blades

The measured bias differed considerably between blades, with six displaying an average negative bias of -3.5%. The remaining two had an average positive bias of 3.7%. Sample Rossi-Alpha plots shown in Figure 11 and Table 5 shows the results. Pre-delay was set to 15µs and gate width was set to 64µs.

The plot above shows the dead time of a blade as being 10µs. This is the processing time of a neutron event, meaning that no two events closer than 10µs can be reported with TTL pulses. Note that this is internal to the blade, so another blade could detect a second neutron immediately after the first.

This sets a limit on pre-delay and therefore forces a lower gate fraction than for a system with a shorter dead time. For the following measurements, pre-delay is set to 15µs to exclude this region.

A count rate of 13000cps in a single blade is equivalent to a 58.5ng ^{252}Cf source in the sample chamber.

The range of biases observed is due to two effects: Undershoot in the analogue electronics within the blade, and the long decay components present in ZnS(Ag). The former induces a negative bias due to a negative undershoot in the pre-amplified signal prior to processing. This undershoot follows every neutron event and lasts for up to 150µs, so any subsequent neutron event arriving during that time will appear to the processing electronics to be weaker than it is. This manifests as a temporary reduction in neutron sensitivity following a neutron event, and a negative bias. The extent of the undershoot depends on specific component values in the analogue electronics and variations between blades are due to tolerances.

The effect of the long decay components of ZnS(Ag) is that each neutron event has a long tail that is visible above

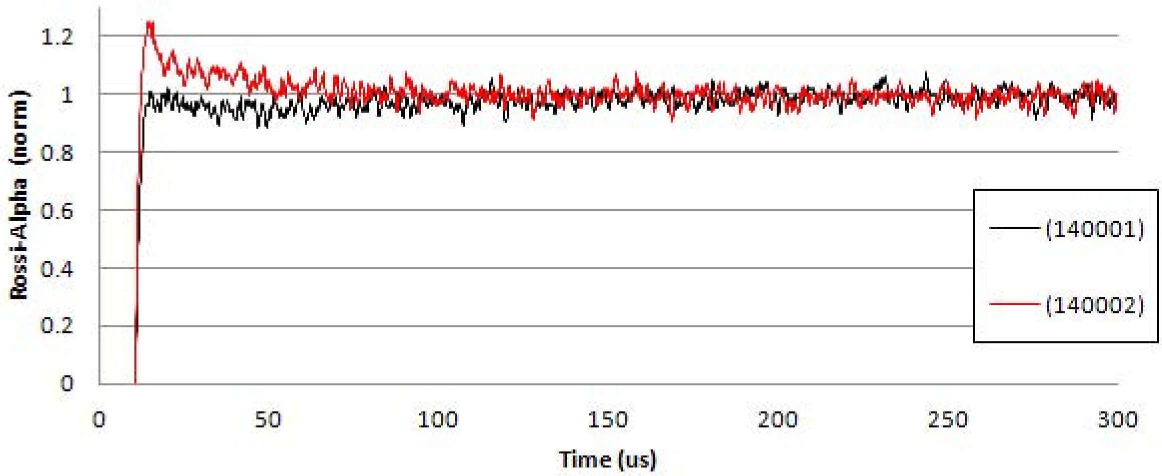


Figure 11: Sample Rossi-Alpha plots with two blades demonstrating the negative bias and positive bias cases. Both are normalised to the average value they settle at beyond 300µs.

S/N	140001	140002	140003	140004	140005	140006	140007	140008
Bias (1800cps)	-4.2%	4.9%	-3.4%	-3.7%	-4.8%	2.6%	-2.5%	-2.6%
Bias (13000cps)	-4.0%	1.9%	-2.4%	-3.3%	-3.4%	2.1%	-1.8%	-2.2%

Table 5: Measured bias for each blade using pre-delay = 15µs and gate width = 64µs.

SiPM noise up to ~100μs after the onset. Subsequent neutron events occurring within that time will be stood on a "pedestal" set by the tail of the first, which increases its signal strength when processed, increasing its chance of passing the discrimination threshold. This manifests as a temporary increase in readout efficiency and thus a positive bias but this effect competes with the negative undershoot. The resultant bias in any given blade is therefore determined by one of these effects dominating the other.

The data appears to show a reduction in bias as count rate increases, whether positive or negative. The mechanism of this requires further study.

5.1.2 Partially Populated NCC

The bias of the NCC with all eight blades was measured with Am-Li sources again at multiple count rates. Blades and HDPE blanks were arranged in Configuration A. Three rates were tested with Am-Li sources to determine whether the total bias changed with count rate. Three different configurations were tested: all blades connected, two positive-bias blades disconnected, two negative-bias blades disconnected.

	All blades	Without positive blades 140002 & 140006	Without negative blades 140004 & 140008
5,000 cps	0.00%	-0.37%	0.00%
14,000 cps	-0.33%	-0.46%	-0.31%
100,000 cps	-0.19%	-0.45%	-0.19%

Table 6: The measured bias of the whole system in three different blade configurations.

We can see that when the whole set of blades is present that the bias is slightly negative and that there is no clear relationship with count rate. Figure 12 shows the normalised Rossi-Alpha plots for the all blades case at the three different count rates. Note how in the region below 10μs the neutron response does not fall to zero, instead falling to 7/8. This is because during that time only the one blade that detected the first neutron is in its dead state and subsequent neutrons can be detected by the other seven blades.

A bias of -0.3% is not considered to be a serious problem since it does not vary with count rate, so it will be accounted for in any plutonium mass measurement calibration (see Section 6).

Attempting to achieve zero bias will be a challenge, since whilst the analogue electronics can be changed to remove the overshoot, that will leave a significant positive bias. This is driven by the long decay time of the ZnS(Ag) which presents a fundamental limit unless the scintillator is changed. This is an area of active research, with some focus on so-called "nickel killed" ZnS which suppresses the long pulses. Other effort is focussed on changing the PSD algorithm to better reject the effect of neutron pile-up described above.

5.2 Selection of Operating Parameters

The pre-delay and gate width parameters are important coincidence settings since they determine the measurement performance of the counter, such as the doubles rate counting efficiency, as well as the relative uncertainty and measurement bias in the doubles rates. Post-processing of stored timed interval distributions was done for varying combinations of pre-delay (4.5, 8, 10, 15, 20, 30 μs) and gate width (16, 32, 64, 128μs) to determine the optimal combination that would minimize the measurement bias and the relative uncertainty in doubles rates, while achieving a gate fraction (fraction of coincidence signal measured) comparable to typical ^{3}He systems. For an

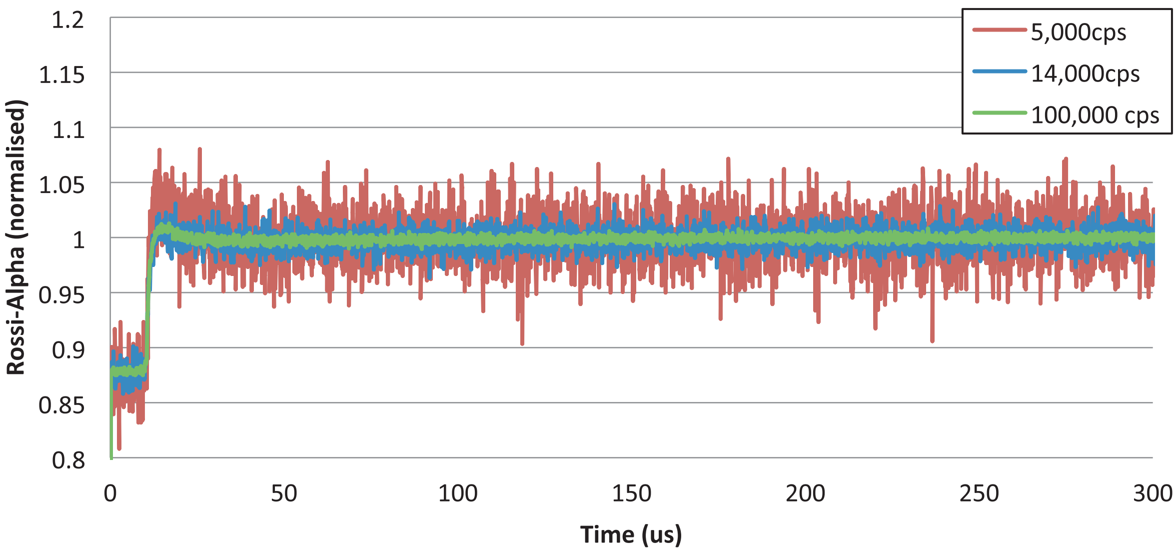


Figure 12: Rossi-Alpha plots taken using all eight blades at three different count rates.

uncorrelated source such as Am-Li the counting rates in the R+A gate should be equal to that in the A gate. If they are not equal, then there is a measurement bias in the counter, likely due to dead time effects and as well as electronics.

Due to the 10µs processing time in each blade, there is a stronger negative bias in doubles for PD<10µs due to the loss of a fraction of the coincidence signal from same-channel correlations. Figure 13 shows the negative bias as a function of pre-delay and gate width for an intense Am-Be source (singles rate of 81,877 cps). Beyond the dead-time effect (i.e. for PD >10us), the observed bias reduces to and stabilizes to negative 0.2%-0.3%, for all three gate width settings. A pre-delay of at least 10us is recommended in this case to minimize bias effects.

Figure 14 shows the relative uncertainty in the doubles rate from a ^{252}Cf source vs. gate width for various pre-delay settings.

The minimum uncertainty is a shallow, flat response between 64µs and 80µs. This corroborates the standard rule-of-thumb for optimal gate width of 1.257 times the die-away time (t)(for an accidentals-dominated thermal system), where, in our case, is t=56.9µs for the 8-blade system, resulting in an optimal gate width of 71.1µs. The fully-instrumented system is expected to have a shorter die-away time of 31.6µs, so the gate width may be shortened, keeping in mind the effect of the gate width on the gate fraction of the counter. A typical HLNCC-II has a typical gate fraction of 0.696.

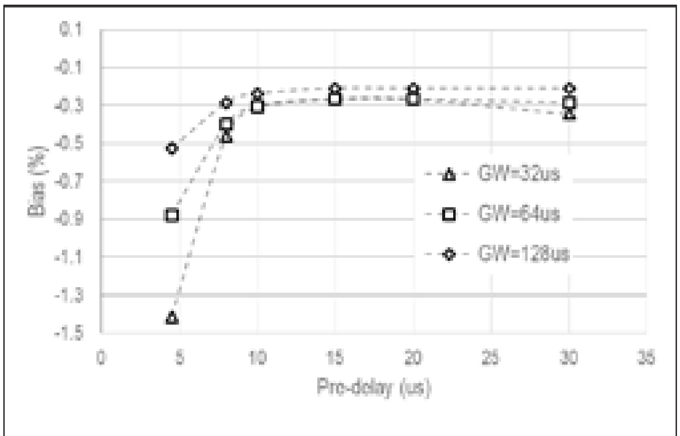


Figure 13: Bias vs pre-delay for fixed gate width values. Am-Li was used in this case.

The values in Table 7 show several choices highlighted in blue from our system that meet or exceed this gate fraction for various pre-delay and gate width combinations. There is no optimum single pair of pre-delay and gate width; rather the choice will ultimately depend on sample type and the precision required for the sample type measurement. For high-alpha samples, it is more important to limit the gate width so as to the limit the accidental contributions at the expense of the gate factor. For the purposes of this work, a 10 µs pre-delay and a gate width of 80µs may be used in the analysis, and results in a gate fraction of 0.633. Despite the larger pre-delay setting of the ^{6}Li-based counter, gate fractions comparable to the HLNCC-II are achievable.

		8-blades	32-blades
PD	GW	t =58.9	t =31.6
10	48	0.478	0.569
10	64	0.566	0.633
10	80	0.633	0.671
10	96	0.684	0.694
8	80	0.656	0.715
8	96	0.708	0.739
7	64	0.597	0.696

Table 7: Gate fractions for pre-delay and gate width combinations. Values with 32 blades are predicted based on the performance estimated in Section 4.

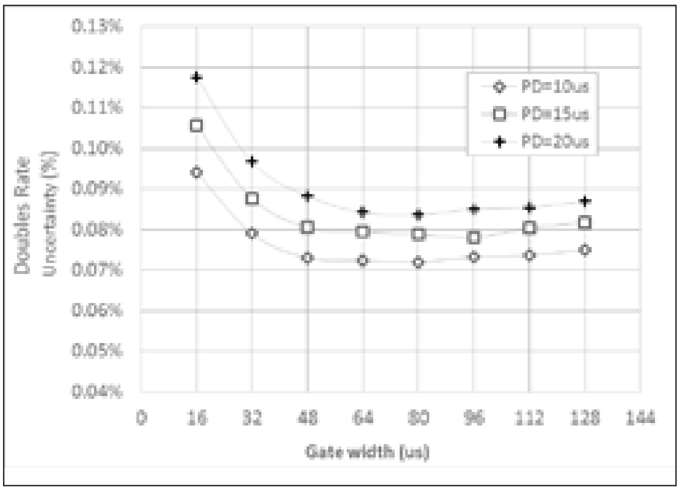


Figure 14: Relative uncertainty in doubles rate vs. gate width

5.3 Gamma-Ray Rejection

Gamma-ray rejection tests of the whole system were carried out using ^{137}Cs and ^{241}Am, two isotopes that the NCC will encounter in service. The objective was to quantify whether the calibration of individual blades to a GRR [6] of better than 10^{-7} was sufficient and whether the blades had a response to the low energy gamma-rays from ^{241}Am.

5.3.1 ^{137}Cs Response

The ^{137}Cs response was tested by placing combinations of ^{137}Cs sources and ^{252}Cf sources into the sample chamber and measuring the total count rate and the doubles count rate. The background-subtracted total count rate when exposed to just ^{137}Cs gives the gamma-ray rejection of the system. Comparing the measured total count rate and doubles rate with ^{252}Cf with and without ^{137}Cs exposure gives the practical effect of that GRR on coincidence measurements.

GRR is calculated as: $GRR = \frac{CR - Background}{A \cdot BR \cdot \Omega / 4\pi}$

Where CR is the measured total rate with ^{137}Cs, A is the source activity in Bq, BR is the branching ratio of the 662keV line (0.85) and Ω is the solid angle subtended by the detector. In this case Ω = 4π.

We have also measured GARRn (gamma absolute rejection ration (neutrons)) which quantifies any change in neutron efficiency due to gamma-ray exposure. It is calculated as:

$$GARRn = \frac{\varepsilon_{abs\,neutron}}{\varepsilon_{abs\,neutron+gamma}}.$$

In this system, it quantifies the effect of gamma-rays piling up with weaker neutron events in a blade to increase the readout efficiency. In all cases, the sources were placed in the centre of the cavity and the eight blades were distributed evenly about the counter in Configuration D. Pre-delay was set to 15μs and gate width was set to 128μs. Table 8 shows the results of these measurements.

We can see that the GRR of the whole system is very low over a broad range of ^{137}Cs activities, never exceeding 10^{-7}. This implies that there is no significant pile-up of gamma-rays in the neutron detector blades.

We can also see that GARRn reaches up to +0.83% when ^{137}Cs during high doses, which is still quite low. It is believed that gamma-rays are adding to the signal of weaker neutron events, causing them to meet the detection threshold. Therefore, this GARRn would quantify an increase in readout efficiency due to neutron-gamma pile-up. More significantly to coincidence counting, we can see that the doubles count rate increases by up to +2.8% against a 1σ uncertainty of ±1.7%. This means that very strong gamma-ray exposure can induce a positive bias in the NCC. Further characterisation of the system is needed to quantify this relationship so that a correction factor with dose can be calculated. Otherwise, the gamma-ray rejection can be improved further by adjusting the neutron detection threshold at the expense of absolute efficiency.

5.3.2 ^{241}Am Response

The response to ^{241}Am was measured using an Am-Li source and an Am-Be source. In both cases, the source was placed at the centre of the cavity, and the total count rate, doubles rate and bias recorded. This was done both with and without a 3mm lead shield. Comparing results gives the response of the system to ^{241}Am gamma-rays. Pre-delay was set to 15μs and gate width was set to 128μs.

Gamma-ray source	Neutron Source	Excess counts due to Cs above background	Change in total rate due to Cs (GARRn)	Doubles rate (Cf only)	Doubles rate (Cf + Cs)
^{137}Cs (3.7MBq)	^{252}Cf	Not measured	+0.28%	105.25 ± 1.25 cps	105.06 ± 1.25 cps (-0.2%)
^{137}Cs (7.4MBq)	^{252}Cf	0.21 cps GRR = 3.3 x 10^{-8}	+0.48%	105.25 ± 1.25 cps	107.7 ± 1.26 cps (+2.3%)
^{137}Cs (370MBq)	^{252}Cf	0.37 cps GRR = 2.1 x 10^{-9}	+0.83%	125.52 ± 1.38 cps	129.07 ± 1.39 cps (+2.8%)

Table 8: Results of measurements with ^{137}Cs and ^{252}Cf. Quoted uncertainties are at the 1σ level.

		Am-Li (S/N 252)	Am-Be (S/N 307)
Total count rate	Shield on	12343.1 ± 4.5 cps	81877.2 ± 11.7 cps
	Shield off	12457.3 ± 8.0 cps	81843.4 ± 11.7 cps
	Change due to Am exposure =	+1.65%	-0.48%
Doubles rate	Shield on	-58.7 ± 7.9 cps	-1794.7 ± 48.3 cps
	Shield off	-59.7 ± 8.0 cps	-1881.4 ± 48.0 cps
	Change due to Am exposure =	+1.62%	+4.83%
Bias	Shield on	-0.32%	-0.21%
	Shield off	-0.31%	-0.22%

Table 9: Results of measurements with Am-Li and Am-Be

Table 9 show that ultimately the bias in the detector does not vary with ^{241}Am gamma-ray exposure, even though there are variations in the measured total count rate and doubles rate.

In the case of Am-Li removing the shield causes a slight increase in totals rate, which is reflected in the slight reduction in negative bias. There is no statistically significant change in the doubles rate. This implies that the blades are sensitive to Am^{241} gamma-rays and that there is a need to either recalibrate the blades to improve gamma-ray rejection or to add a liner to the sample cavity. The Am-Be case is different in that the singles rate does not change much when ^{241}Am gamma-rays are introduced. We also observe a noticeable decrease in the negative doubles rate when the shield is introduced. This is to be investigated further.

5.4 Absolute Efficiency Profile

The absolute efficiency of the NCC was profiled in two sets of measurements. The first was a vertical line up the centre of the cavity, and the second was a grid of points in a horizontal plane at the mid-height of the cavity. In all positions a pre-delay of 15μs and a gate width of 64μs. The blades and HDPE blanks were arranged in Configuration C. In all cases, a bare ^{252}Cf source of 25.5ng (59000 n/s) was used.

5.4.1 Vertical Profile

The vertical profile plot shows that the absolute efficiency varies quite strongly with vertical position and that the doubles rate drops to as much as 73% that at the centre. This is consistent with the square of the drop in absolute efficiency indicating that the absolute efficiency profile dominates the doubles rate profile and there are no significant effects due to gate fraction. This can be confirmed by Figure 16 which shows the measured die-away time as a function of vertical position. It is quite consistent and appears to be slightly lower at the extremes, so the gate fraction would not reduce at those points.

Absolute efficiency is lower at the bottom of the chamber indicating that perhaps the blades have lower sensitivity at their ends, warranting further investigation. Until this problem is resolved, testing will be limited to the centre of the sample cavity, and as it stands, the NCC is unsuitable for measuring extended sources.

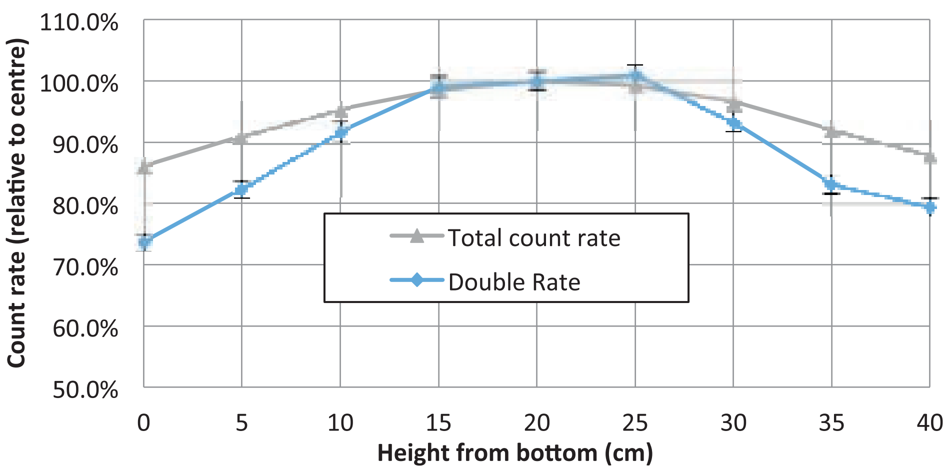


Figure 15: Plots of how the total count rate and doubles rate vary with vertical position.

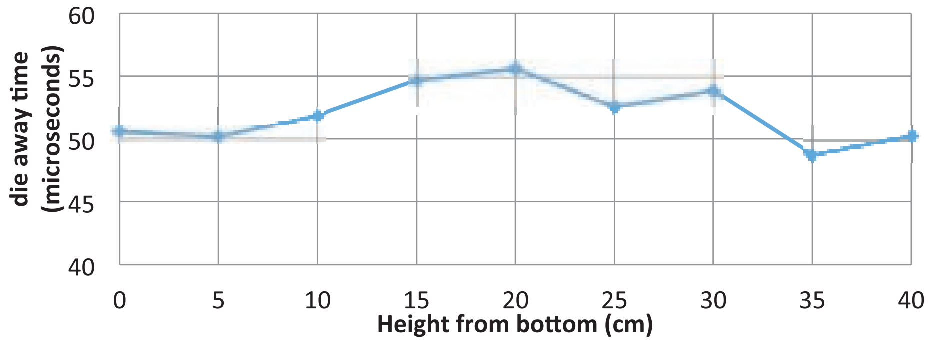


Figure 16: Die-away time as a function of vertical position. 20cm represents the centre of the sample cavity.

5.4.2 Horizontal Profile

Measurements of absolute efficiency, doubles rate and die-away time were taken over a horizontal plane at the mid-height of the sample cavity. Results are shown in Figure 17 and Figure 18, expressed as values relative to the centre position.

It is clear that with the blades and HDPE blanks in Configuration C, the NCC exhibits poor horizontal uniformity, with clear peaks at the sides and nadirs and the corners. This is explained by the concentration of blades and extra HDPE in the middle of the slabs, and away from the corners of the NCC.

Totals rate	X = -8.0	-7.5	-5.7	-4.0	-2.8	0.0	2.8	4.0	5.7	7.5	X = 8.0
Y = -8.0						105%					
-7.5		96%								97%	
-5.7			100%						101%		
-4.0						100%					
-2.8					100%		101%				
0.0	107%			101%		100%		102%			108%
2.8					101%		101%				
4.0						102%					
5.7			101%						102%		
7.5		98%								98%	
Y = 8.0						108%					

Figure 17: A plot of how the total count rate varies over the horizontal plane relative to the central position.

Doubles rate	X = -8.0	-7.5	-5.7	-4.0	-2.8	0.0	2.8	4.0	5.7	7.5	X = 8.0
Y = -8.0						114%					
-7.5		91%								96%	
-5.7			100%						101%		
-4.0						105%					
-2.8					101%		102%				
0.0	113%			99%		100%		103%			118%
2.8					103%		105%				
4.0						103%					
5.7			104%						104%		
7.5		96%								99%	
Y = 8.0						120%					

Figure 18: A plot of how the measured doubles rate varied over the horizontal plane. All values are given relative to the central position. Pre-delay was set to 15µs and gate width was set to 64µs.

The doubles rate shows a great non-uniformity that follows the same pattern as the total count rate, as expected. As in the vertical profile case, the measured non-uniformity of the doubles rate agrees well with the square of the non-uniformity in totals count rate. This implies that the gate fraction does not significantly vary over the horizontal plane, which would otherwise cause the doubles rate to vary independently of the total count rate.

There appears to be a bias towards higher total count rate and doubles rate in the +X and +Y directions. Such observations can be explained by the fact that the measurements were not taken in a low-scatter environment. In both the +X and +Y directions scattering material was present which would return partially moderated neutrons above the cadmium cut-off energy which would then thermalise inside the NCC.

6. Assay of Pu Samples

The final performance parameter of interest, and one that relies on those explored above, is the linearity of the NCC plutonium mass calibration function. To measure this, a number of plutonium-gallium (Pu-Ga) samples were tested at JRC. They were all taken from the same series and this have the same isotopic composition, shown in Table 10. The samples were thin disks and were placed at the centre of the sample cavity on aluminium stands.

Analysis was carried out using the PTR-32 and INCC software to extract the effective plutonium mass. As according to the analysis in Section 5.2, the pre-delay was set to 10µs and the gate width was set to 80µs. Figure 19 shows the results in terms of doubles rate as a function of known plutonium mass.

Isotope	Iso.Compo. wt %	rsd %	Specific Power mW/g (error)	Half Life (y)
Pu-238	0.1336	0.04	567.57 (0.26)	87.74
Pu-239	75.6606	0.03	1.9288 (0.0003)	24119
Pu-240	21.4898	0.07	7.0824 (0.002	6564
Pu-241	1.9510	0.93	3.412 (0.002)	14.348
Pu-242	0.7651	0.38	0.1159 (0.0003)	376300
Am-241	1.86	0.02	114.2 (0.42)	433.6

Table 10: The isotopic composition of the Pu-Ga samples tested here.

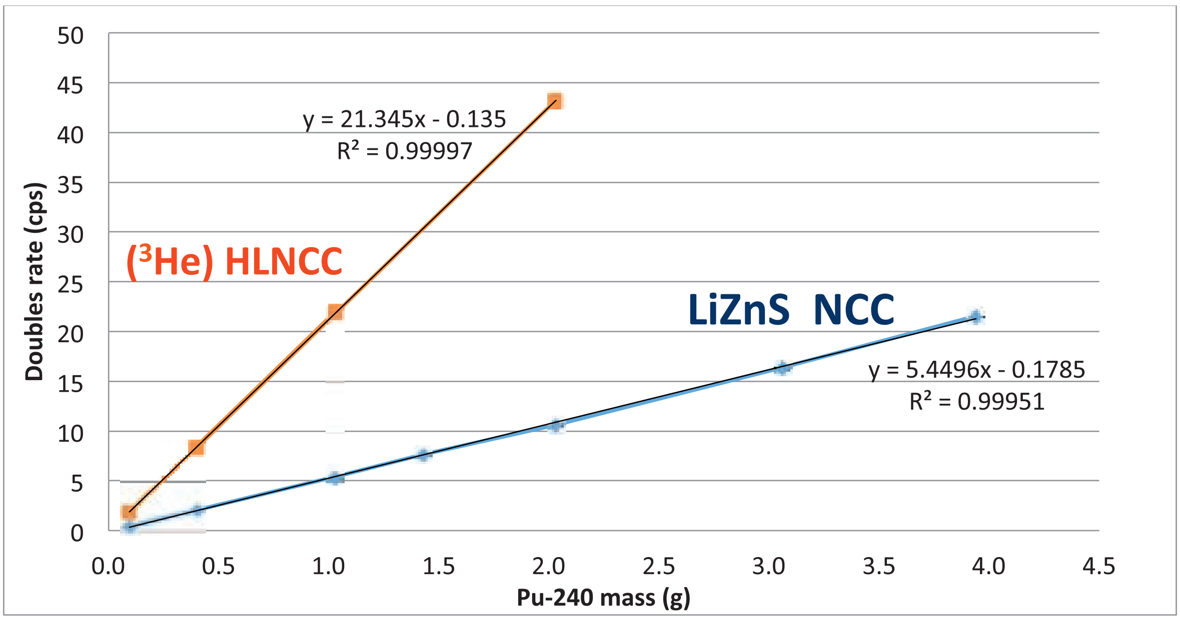


Figure 19: A plot of doubles rate as a function of Pu mass for the partially populated NCC and an HLNCC-II.

The NCC was set up in Configuration C with two blades and two blanks per module. For comparison, the same plutonium samples were measure by an HLNCC-II also at JRC Ispra.

We can see a linear relationship between plutonium mass and measured doubles rate. The linear fit shows a negative intercept, which reflects the negative bias measured in this NCC. The HLNCC-II shows a far greater gradient due to its higher efficiency (16.5% vs 9%), and a much smaller negative intercept since it has nearly zero bias.

7. Conclusion

A prototype NCC was designed and tested in a partially populated configuration against a benchmark set by a HLNCC-II. The results of that testing were then used to predict the performance of a fully-populated NCC with encouraging results. In summary, our findings were:

- The readout efficiency of the blades is high at 85.1% and their gamma-ray rejection is excellent.
- The FoM of the partially-populated NCC is lower than the HLNCC-II at ~1.3 depending on configuration, but this is sufficient to obtain good data.
- The gamma-ray rejection of the whole NCC is good but it shows a slight sensitivity to very high gamma-ray fluxes due to neutron-gamma pile-up
- The NCC shows a negative bias of -0.3% due to the analogue electronics in the blade. This can be developed further but achieving zero bias is limited by the length of the ZnS(Ag) scintillation pulse.
- The profile of the partially-populated NCC is poor. Whilst this is expected in the horizontal plane, it was not in the vertical plane. This restricts plutonium mass measurements to point sources at the centre of the cavity.
- Plutonium mass measurements show a very linear relationship.

This performance is deemed to be very good in a prototype though some work remains to solve the problems that have been highlighted.

8. Future Work

Now that a reduced version of a neutron coincidence counter has been characterized and calibrated, further work can be considered.

Firstly, the counter will be fully instrumented with 32 blades and then characterized and calibrated in the same way. Furthermore, recent improvements in SiPM technology will be incorporated in those blades to improve dead time and thermal neutron detection efficiency further. The optical design of the blades will be investigated to quantify any loss of readout efficiency along their length.

Other coincidence counter geometries can be explored which are predicted to offer better absolute efficiency profiles, especially in the vertical plane. Another improvement to the coincidence counter will be the addition of a low-Z liner to aid in rejection of low energy gamma-rays.

The positive bias induced by the long decay time of the ZnS(Ag) scintillator will be addressed by investigating alternative scintillators such as ZnS(Ag,Ni) or “nickel killed” ZnS which shows suppression of the long components. Another avenue of research is changing the PSD algorithm to better detect the strength of the tail and remove it from the measurement of neutron event strength, to remove this effect.

A range of high-rate samples will be tested to provide empirical data of the systems dead time behavior. This data will be used to compute a correction matrix.

On the broader scale, the modules developed for this counter can be used in an active system as an UNCL. The blades are also suitable for development of a module for multiplicity counters such as the ENMC, especially if their dead time can be reduced to exploit the low die-away times of such systems. If not, then any ENMC can be expected to contain a very high number of blades (~200) so the step due to dead time will only introduce a bias of -0.5%.

9. Acknowledgements

The authors would like to thank DG-ENER for supporting the project. We also thank the staff at JRC Ispra for their ready assistance in taking measurements, and the engineers at Symetrica Security for their excellent work in producing the system itself.

10. Legal matters

10.1 Privacy regulations and protection of personal data

The authors agree that ESARDA may print their names/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.

10.2 Copyright

11. References

[1] Menlove HO, Krick, M. (1979). The High-Level Neutron Coincidence Counter (HLNCC): User's Manual. Report LA-7779-M. Los Alamos, NM: Los Alamos National Laboratory.

[2] Kouzes RT. 2009. The ^{3}He Supply Problem. Technical Report PNNL-18388, Pacific Northwest National Laboratory, Richland, WA.

[3] G. Dermody, A Scalable 3He---free neutron detector for multiple applications. Presentation at the ESARDA Novel Approaches/Novel Technologies Working Group workshop held in Oxford in March 2014

[4] Current Status of ^{3}He Alternative Technologies for Nuclear Safeguards, international measurement campaign and workshop on He3 alternatives for safeguards, at the JRC in Ispra October 2014, within a DOE-Euratom Action Sheet 47. Report LA-UR-15-21201 Ver. 2, 2015

[5] P. Peerani, C. Carrapico, B. Pedersen, V. Forcina, F. Rosas, A. Rosite, H. Tagziria, A. Tomanin, Workshop on He-3 alternatives for safeguards applications, presented at the 37th ESARDA symposium in Manchester May 2015

[6] Kouzes RT, JH Ely, AT Lintereur, EK Mace, DL Stephens, ML Woodring. 2011. Neutron detection gamma ray sensitivity criteria. NIM A 654 (2011) 412–416.

Detection of fission signatures induced by a low-energy neutron source

A. Ocherashvili[a], V. Mayorov[b], A. Beck[a], G. Heger[c], E. Roesgen[b], J.-M. Crochemore[b], M. Mosconi[b], B. Pedersen[b]

[a] Physics Department, Nuclear ResearchCenterNegev, P.O. Box 9001, 4190 Beer-Sheva, Israel
[b] Nuclear Security Unit, Institute for Transuranium Elements (ITU), Joint Research Centre, European Commission, Via E. Fermi 2749, Ispra 21027 (VA), Italy, bent.pedersen@jrc.ec.europa.eu
[c] Israel Atomic Energy Comission, P.O. Box 7061, 61070 Tel Aviv, Israel

Abstract:

We present a method for the detection of special nuclear materials (SNM) in shielded containers which is both sensitive and applicable under field conditions. The method uses an external pulsed neutron source to induce fission in SNM and subsequent detection of the fast prompt fission neutrons. The detectors surrounding the container under investigation are liquid scintillation detectors able to distinguish gamma rays from fast neutrons by means of the pulse shape discrimination method (PSD). One advantage of these detectors, besides the ability for 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 the detection of the prompt fission neutrons in multiple detectors while benefiting from a low accidental (background) coincidence rate yielding a low detection limit. Another principle advantage of this method derives from the fact that the external neutron source is pulsed. By proper time gating the interrogation can be conducted by epithermal and thermal source neutrons only. These source neutrons do not appear in the fast neutron signal following the PSD analysis thus providing a fundamental method for separating the interrogating source neutrons from the sample response in form of fast fission neutrons. The paper describes laboratory tests with a configuration of eight detectors in the Pulsed Neutron Interrogation Test Assembly (PUNITA). The sensitivity of the coincidence signal to fissile mass is investigated for different sample configurations and interrogation regimes.

Keywords: Nuclear security, SNM detection, PSD, neutron generator

1. Introduction

Passive and active non-destructive assay (NDA) methods have potential in practical applications as a means to detect special nuclear materials (SNM). The prompt emission from fission of neutrons and γ-rays appear to be useful signatures for the detection of SNM in shielded containers. One reason for this is that a component of the prompt γ-rays from fission are of high energy and thus very penetrating and 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 γ-rays and neutrons are emitted simultaneously [1-3].

Using an external neutron source to induce fission extends the usefulness of this detection method to apply not only to spontaneous fissile elements but also to elements 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 (epi-thermal or thermal) neutron flux only, providing the possibility to distinguish the fast fission neutrons from the low energy source neutrons in the neutron detection system [4].

In the present work we study epi-thermal neutron interrogation only. The thermal interrogation has been demonstrated to yield a response proportional to the fissile mass [4]. In epi-thermal interrogation, which is desirable for nuclear security purposes, the response is harder to interpret due to the large proportion of gamma detection events in the detector during slowing-down of the source neutrons.

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. Figure 1 shows a cross section of PUNITA and the positioning of the detectors used in this work. The facility is composed of a large graphite liner surrounding a central cavity of volume

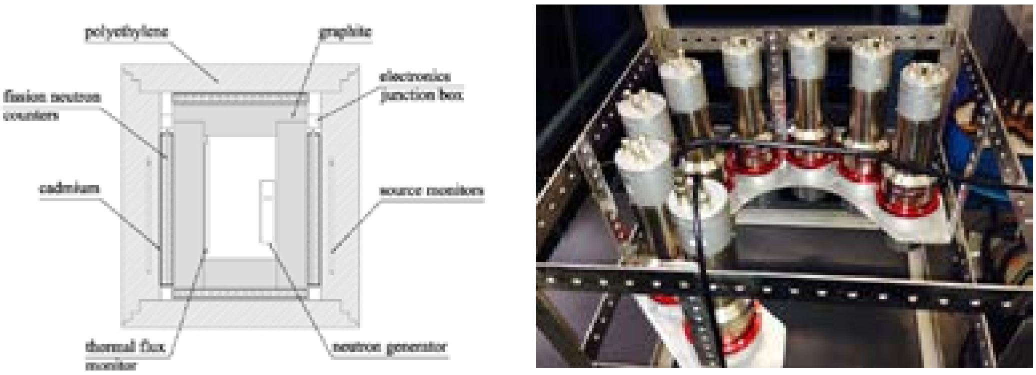


Figure 1: Sketch of PUNITA showing the permanently mounted neutron detectors and the neutron generator mounted inside the sample cavity (left picture). The right hand picture shows the positioning of the eight liquid scintillation detectors within the sample cavity of PUNITA.

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 ^3He neutron detectors are embedded in polyethylene modules and shielded by cadmium (fission neutron counters in Figure 1). In the present experiments these detectors are used as reference detectors of the prompt fission neutrons.

In Figure 1 is also indicated, as source monitors, bare ^3He neutron detectors which are used to normalize detector readings in all experiments to the same total neutron emission from the generator target. The neutron generator (Model A-211 from Thermo Fisher Scientific Inc.) is pulsed at 100 Hz which is chosen based on 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 [6]. The generator is able to produce short and intense bursts of neutrons with 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 120 μs, and a thermal period from 250 μs to 9 ms, respectively [4].

We use an array of eight 3”x3” liquid scintillation detectors EJ-309 from Eljen Technology [5] for the detection of the prompt radiation from fission events (Figure 1). These detectors can distinguish fast neutron interactions from other interactions by means of pulse shape discrimination (PSD). The detection pattern is based on the simple fact that detection of fast fission neutrons is evidence for the presence of fissile material. The performance of scintillation detectors with respect to γ/n discrimination in the PUNITA facility is described in [4]. Due to the very fast response of the scintillation detectors the effect of the neutron generator burst can be followed in detail [7].

The anode output of the photomultiplier is connected directly to a signal digitizer. Each detector was supplied with individual high voltage (NDT1740) [8] to allow having same response in all detectors to a given photon source (^{137}Cs (Eγ=662 keV). The detectors were calibrated using the following photon sources: ^{133}Ba (Eγ=356 keV), ^{137}Cs (Eγ=662 keV), ^{54}Mn (Eγ=835 keV), and ^{22}Na (Eγ=511,1274 keV). The upper end of the dynamic range is set to eliminatethe 2.223 MeV photons producedby thermal neutron capture in hydrogen, and the lower end of the dynamic range is set to the PSD resolution value at 120.6 keVeeas achieved with a ^{252}Cf source.

The signal digitizers used in this work are from Signal Processing Devices Sweden AB (http://spdevices.com/). Figure 2 shows the triggering and data processing scheme used in these experiments.

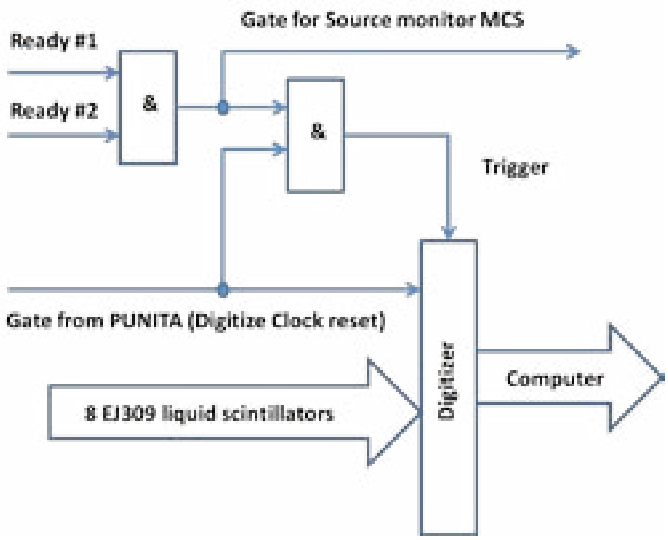


Figure 2: Data triggering scheme.

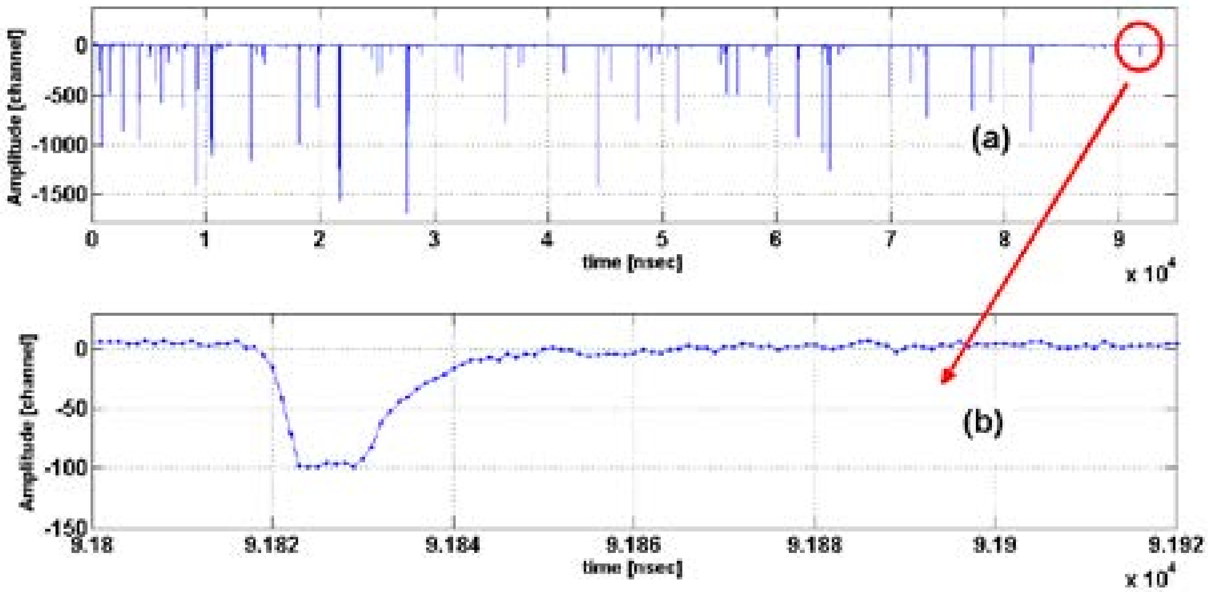


Figure 3: Typical waveform of epithermal data (a), in picture (b) is shown analyzes of the last signal. Time zero in picture (a) corresponds to 28µs after the 14-MeV burst.

By delaying the data recording period from the neutron generator burst (Figure 2, "Gate from PUNITA") the data acquisition can be tailored to a certain neutron energy range. In an earlier work interrogation was done by thermal neutrons only using a triggering scheme based on detection of multiple signals [7]. In the present work concerned with epi-thermal neutron interrogation this triggering scheme is not efficient due to the very high rate of photon detections during the slowing down of the generator neutrons. In contrast the present triggering scheme (Figure 2) is very simple. A data stream from the eight scintillation detectors is recorded following a "ready" signal from the digitizers and for the duration of the PUNITA Gate. The recorded waveforms are 95 µs long, digitized at 1 GS/s and 12-bit resolution and constitute a single data cycle. Such data streams are recorded at 100 Hz.

All recorded waveforms are analyzed offline in MATLAB [9]. A signal is defined as having an amplitude larger than 3σ of the baseline variation. Examples are shown in Figure 3. All signals are extracted by such criteria.

The initial data analysis produces a list of all signals. The signals are described in terms of: signal amplitude, time stamp (t_0), detector number and PSD value. From this list the neutron signals can be selected as in standard list-mode operation of neutron multiplicity analyzers. This allows us the possibility of using standardized principles from passive neutron multiplicity analyses. In addition we can analyze mixed photon/neutron streams.

3. Measurements of uranium samples in PUNITA

A series of standard CNNM U_3O_8 sources [10] are used in conjunction with the pulsed neutron interrogation and eight EJ-309 scintillation detectors. The five CBNM standards are identical in all aspects (total U mass of about 169 grams, 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 centered among the eight detectors at a distance of 150 mm.

Figure 4 shows the MCNP simulated source neutron spectrum in discrete periods of the range 27 µs to 135 µs after the 14-MeV burst. In the present analysis we use the period 28 µs to 123 µs i.e. 95 µs.

Also shown in Figure 4 are the capture and fission cross-sections of some isotopes. Clearly in the selected time period fission is only induced in the ^{235}U isotope. One can also estimate that for CBNM samples of small ^{235}U content, the capture reaction in ^{238}U becomes relatively important compared to fission in ^{235}U.

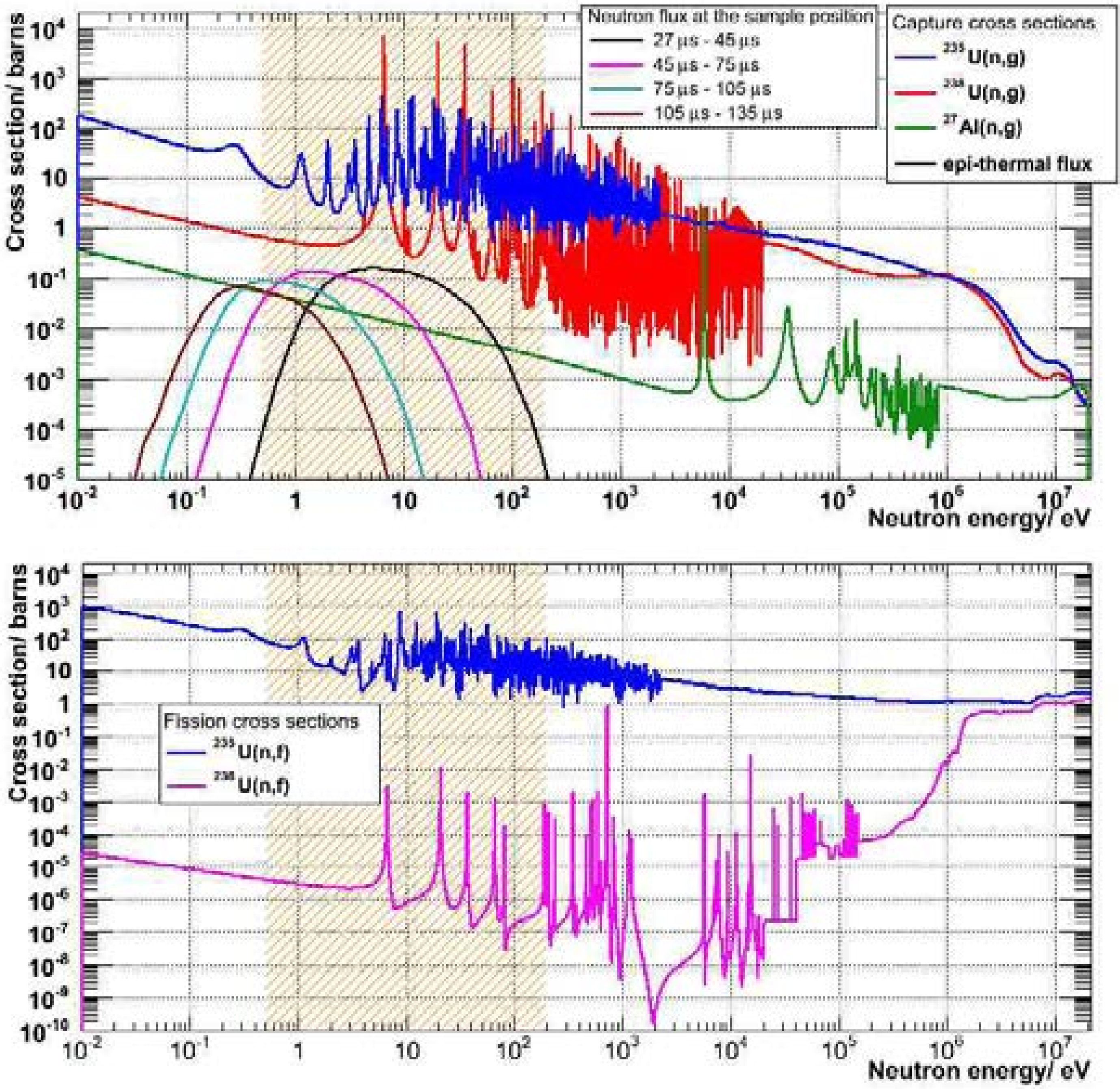


Figure 4: Epi-thermal neutron cross section for ^{235}U(n,γ), ^{238}U(n,γ), and Al((n,γ), (top picture) and ^{235}U(n,f), ^{238}U(n,f) (bottom picture).

3.1 Neutron multiplicity counting

As mentioned above continuous waveforms of 95 µs length were recorded, starting at 28 µs delay after the 14-MeV neutron burst. Figure 5 shows a distribution of the pulse time stamp ($t_{0)}$for neutrons only.

In Figure 5(a) the detected neutron counts decreases until about 30 µs. The neutron detections in this range are mostly fast neutrons from the generator. Neutrons of energy below about 700 keV do no longer produce a signal (PSD) in the detectors associated with neutron detection. After 30 µs most detected neutrons are fast fission neutrons. The slowly falling rate of neutron detections is due to the decaying neutron flux (in spite of the increasing fission cross section for lower energies). We divide our data analyses into three parts: 28-59 µs, 59-81 µs and 81-123 µs. For this data we use a kind of Shift Register analysis on the neutron associated “Rossi-alpha” kind of distributions.

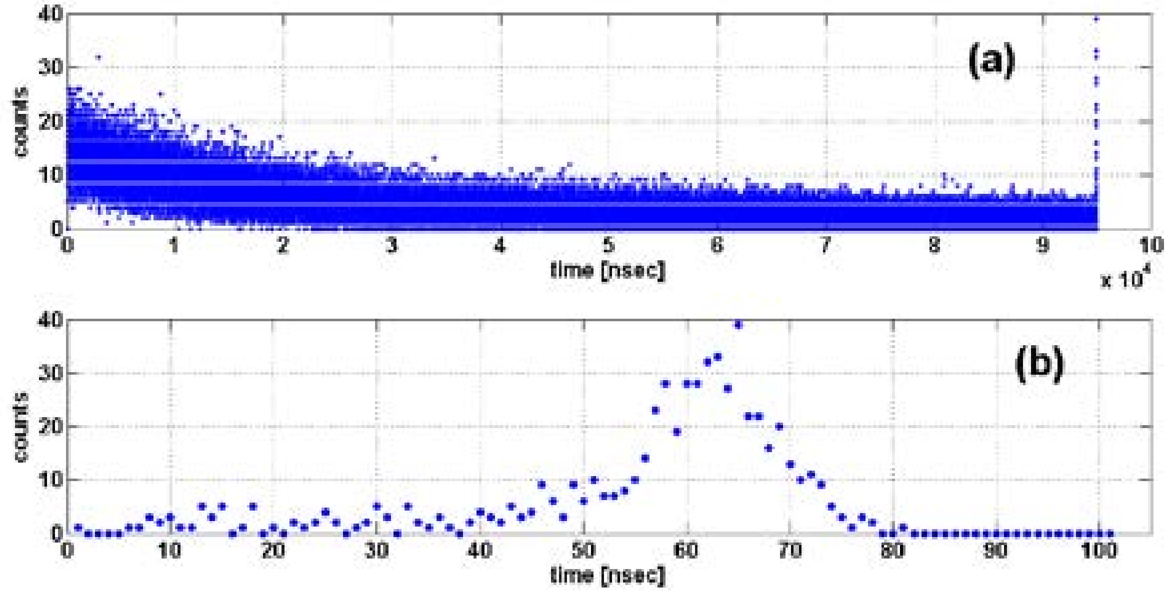


Figure 5: Distribution of time stamps (t_0) for neutron associated pulses (a), picture (b) shows zoom of last 100 ns of the waveform for the CBNM446 sample (4.46% ^{235}U). Time zero in picture (a) corresponds to 28 µs after the 14-MeV burst.

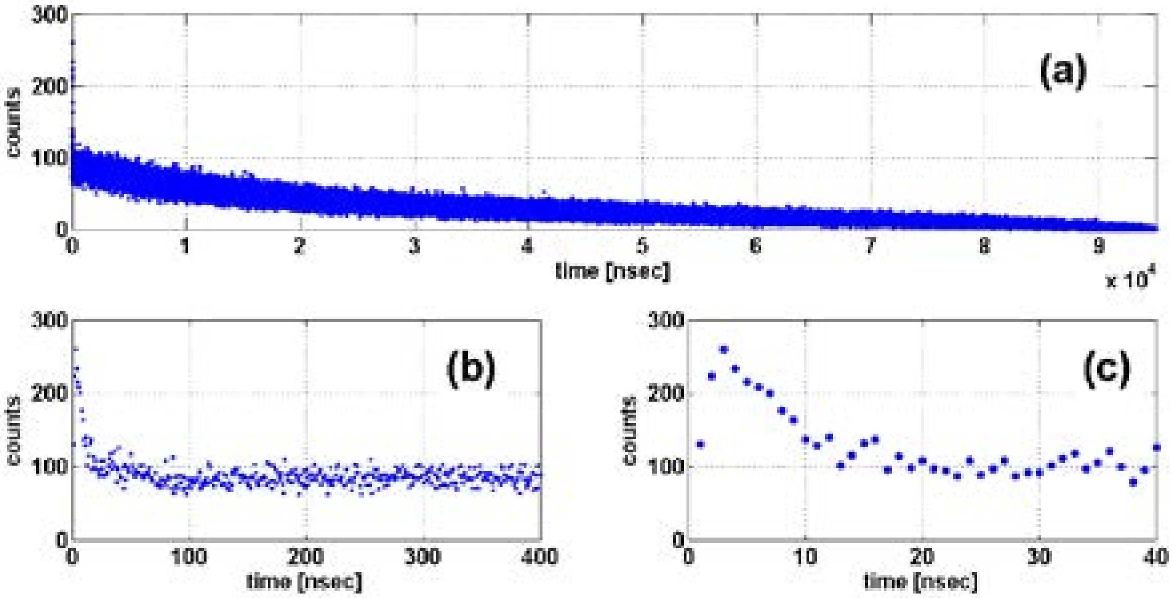


Figure 6: Neutron asosiated "Rossi-alpha" kind of distribution for CBNM446 sample: (a) entire picture, (b) zooming of first 400 ns and (c) zooming of first 40 ns.

As can be seen in Figure 6(b) a "Rossi-alpha" distribution of detected neutrons quite similar to what is observed in standard passive neutron counting of spontaneous fission events although the time scale is quite different. We consider two time gates: one immediately following a neutron signal of length 20 ns (called the prompt gate (PG)), and another in the period 250÷270 ns after the first (called the delayed gate (DG)). We form frequency distributions of number of neutron detections in the gates, calculate the 1st factorial moment from the distribution, and make the subtraction PG – DG. For different samples measurements this result is normalized to the Source Monitor (SM) counts (proportional to the neutron emission from the generator). Figure 7 shows the results of the CBNM uranium samples.

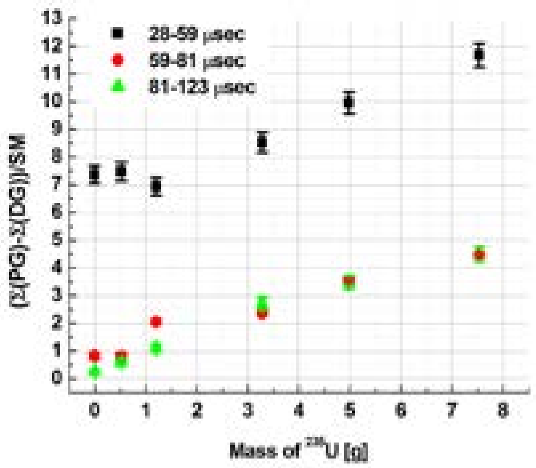


Figure 7: Difference between PG and DG gates of neutron pair events normalised to the neutron emission of the neutron generator.

For the time period of 28-59 µs the linearity is not good for the smallest sample and the empty container. The reason might be that fast source neutrons still persist in this range.

The result for counting the total number of neutron detection events (single neutrons) is presented in Figure 8. In this case the earlier period (28-59 µs) does not show linearity with fissile mass. The reason is likely due to source neutrons still being observed as fast neutrons in the detectors.

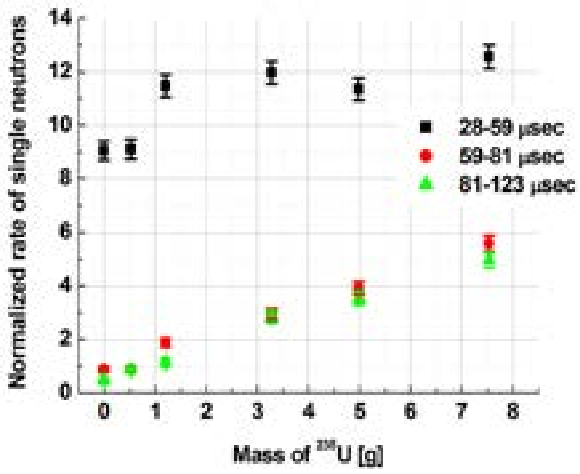


Figure 8: Detected normalized rate of single neutrons.

4. Conclusions

In the present work we have investigated a detection method for special nuclear material based on an epi-thermal source of neutrons inducing fission is fissile isotopes, and the detection of fast prompt fission neutrons as a signature of the presence of fissile material. The advantage of this method is that epi-thermal neutrons have sufficiently low energy not to leave a neutron signature in the liquid scintillation detectors, while the neutron energy is sufficiently high for the neutrons to pass through thermal neutron shielding, and induce fission in fissile isotopes. The suitable source neutron energy range is selected by varying the delay of interrogation following a burst of 14-MeV neutron from a pulsed neutron generator placed in a strongly moderating detection assembly. The difficulty in epi-thermal interrogation is the overwhelming photon response in the detectors during the slowing-down of the source neutrons. By recording all detection events in eight scintillation detectors in a selected period of about 130 µs during slowing-down of the source neutrons, a clear signature, proportional to the fissile mass, of prompt neutrons from induced fission by epi-thermal neutrons was observed.

The purpose of this work is to investigate the feasibility of a device for the detection of SNM that would combine epi-thermal and thermal neutron interrogation, and would be both sensitive to the presence of fissile materials and be able to overcome potential thermal neutron absorbers placed around the fissile material.

5. References

[1] Enqvist A, Flaska M, Dolan J L, Chichester D L, Pozzi S A; *A combined neutron and gamma-ray multiplicity counter based on liquid scintillation detectors;*Nucl. Instr. Meth. A; 652; 2011; p. 48-51.

[2] Pozzi S A, Clarke S D, Flaska M, Peerani P; *Pulse-height distributions of neutron and gamma rays from plutonium-oxide samples*; Nucl. Instr. Meth. A; 608(2); 2009; p. 310-315.

[3] Clarke S D, Flaska M, Pozzi S A, Peerani P; *Neutron and gamma-ray cross-correlation measurements of plutonium oxide powder*; Nucl. Instr. Meth. A; 604(3) 2009; p. 618-623.

[4 Ocherashvili A, Roesgen E, Beck A, Caspi E N, Mosconi M, Crochemore J M, Pedersen B; *SNM detection by means of thermal neutron interrogation and a liquid scintillation detector*; JINST, 7 CO3037, 2012

[5] www.eljentechnology.com

[6] Favalli A, Mehner H-C, Crochemore J-M, Pedersen B; *Pulsed Neutron Facility for Research in Illicit Trafficking and Nuclear Safeguards*; IEEE Transactions on Nuclear Science; 56(3); 2009; p. 1292-1296.

[7] A. Ocherashvili, M. Mosconi, J-M. Crochemore, A. Beck, E. Roesgen, V. Mayorov, B. Pedersen, Fast neutron coincidences from induced fission as a method for detection of SNM, ESARDA Bulletin, Issue No 49, p42-51, July 2013.

[8] CAEN High Voltage unitNDT1740, http://www.caen.it/

[9] MATLAB 7.0 and Statistics Toolbox 7.1, The Math Works, Inc., Natick, Massachusetts, United States.

[10] Carpenter B S, Gramlich J W, Greenberg R R, Machlan L A, DeBievre P, Eschbach H L, Meyer H, Van Andenhove J, Connelly V E, Trahey N M, and Zook A; Standard Reference Materials: Uranium 235 Isotopic Abundance Standard Reference Materials for Gamma Spectrometry Measurements; National Bureau of Standards Special Publication 260-96; U.S. Government Printing office Washington; 1986.

[11] X-5 Monte Carlo Team Diagnostics Applications Group Los Alamos National Laboratory; MCNP – A general Monte Carlo N-Particle Transport Code, Version 5, Volume I: Overview and Theory; LA-UR-03-1987; 2003; Los Alamos National Laboratory

Emerging Applications of Bottom-Up Uncertainty Quantification in Nondestructive Assay

[1]Burr, T., [2]Croft, S., [3]Dale, D., [3]Favalli, A., [3]Weaver, B., [3]Williams, B.

[1]International Atomic Energy Agency, Vienna, Austria
[2] Oak Ridge National Laboratory, Oak Ridge, TN, US
[3] Los Alamos National Laboratory, Los Alamos, NM, US
E-mail: tburr@iaea.org, crofts@ornl.gov djdale@lanl.gov,afavalli@lanl.gov,theguz@lanl.gov, briaw@lanl.gov

Abstract

The Guide to the Expression of Uncertainty in Measurement (GUM) provides guidance on expressing measurement uncertainty for calibration, laboratory accreditation, and metrology services. Nondestructive assay (NDA) of items containing nuclear material uses calibration and modelling to infer item characteristics on the basis of detected radiation such as neutron and gamma emissions. Uncertainty quantification (UQ) can be approached from a bottom-up or top-down analysis. Top-down UQ compares measurements of the same item(s) from multiple assay techniques and/or laboratories. Bottom-up UQ quantifies sources of input uncertainty to a process and the resulting output uncertainty. Bottom up UQ is the focus of this paper. UQ for NDA has always been important, but there is a need for application of better statistical techniques and for UQ to play more of a role in assay development and assessment. This paper describes NDA applications (the enrichment meter, distributed-source term analysis, Cf shuffler, Uranium neutron collar) that have emerging UQ topics that are not specifically addressed by the GUM, including: item-specific biases, errors in predictors, model error effects, and quantification of uncertainty in computer model errors and parameters. This paper also includes an initial UQ case study using the Uranium neutron collar assay method.

Keywords: Bottom-up uncertainty quantification (UQ); emerging UQ topics; errors in predictors; item-specific biases

1. Introduction

Uncertainty quantification (UQ) for non-destructive assay (NDA) in nuclear safeguards applications has always been important. But, currently it is recognized that greater rigor is needed and achievable using modern statistical methods and by letting UQ have a more prominent role in assay development and assessment [1-3]. UQ is often difficult, but if done well, can lead to improving the assay procedure itself. Therefore, we describe the extent to which the guideline for expression of uncertainty in measurements (GUM) [4] can be used for UQ in NDA [1-3]. This paper also takes steps toward better UQ for NDA by illustrating UQ challenges that are not addressed by the GUM. These challenges include item-specific biases, calibration with errors in predictors, and model error, especially when the model is a key step in the assay. We briefly describe a specific NDA application, the enrichment meter principle, for which a variation of the GUM approach can be applied, and other NDA applications for which the GUM approach must be extended. Then, a case study of the Uranium neutron collar is presented. The paper is organized as follows. Section 2 gives additional background on NDA and UQ for NDA. Section 3 describes the GUM and a few example NDA applications for which the GUM is applicable. Section 4 is the Uranium neutron collar (UNCL) case study. Section 5 is a discussion and summary.

2. UQ for NDA

The GUM [4] indirectly addresses top-down methods, but is most known for illustrating a bottom-up option that applies uncertainty propagation of uncertainty in each assay step to estimate the uncertainty in the assay. For bottom-up UQ, the GUM's measurement equation is

$$Y = f(X_1, X_2, ..., X_p) \qquad (1),$$

where Y is the estimate of the measurand, and the Xs are p inputs. The p inputs can be measurement or adjustment factors, and can be regarded as having a joint probability distribution that can include covariances among some or all of the inputs. For example, some of the inputs can be estimated calibration parameters, others can be measured values, and others can be adjustment factors. The GUM distinguishes two types of evaluations to describe how to estimate the probability distribution of one or more inputs. "Type A" evaluations uses statistical methods applied to measured data. "Type B" evaluations use judgments, models, or other non-statistical methods. The inputs Xs in Eq. (1) can be obtained using Type A and/or Type B evaluations.

There is recent interest in revising and extending the GUM for reasons described in Bich [5]. However, there are many applications for which Eq. (1) is adequate for defensible UQ for Y. One needs to know the functional form $f()$, and know how to quantify the error magnitudes in each of the Xs. Because the GUM does not focus on top-down UQ, there is no attempt to describe sources of uncertainty such as omitted physical effects or model uncertainty.

In NDA applications, items emit neutrons and/or gamma-rays that provide information about the source material, such as isotopic content. However, item properties such as density, or the distribution of radiation-absorbing isotopes, which relate to neutron and/or gamma absorption behaviour of the item, can partially obscure the relation between detected radiation and the source material; this adds a source of uncertainty to the estimated amount of special nuclear material (SNM) in the item. One can express item-specific impacts on uncertainty using a model such as

$$CR/M = g(X_1, X_2, ..., X_p) \qquad (2),$$

where CR is the item's neutron or gamma count rate, M is the item SNM mass, g is a known function, and $X_1, X_2, ..., X_p$ are p auxiliary predictor variables such as item density, source SNM heterogeneity within the item, and container thickness, which will generally be estimated or measured with error and so are regarded as random variables [6,7].

To map Eq. (2), to GUM's Eq. (1), write

$$M = CR / g(X_1, X_2, ..., X_p) = h(X_1, X_2, ..., X_{p+1}) \quad (3),$$

where the measured CR is now among the p+1 inputs. Because p is generic notation for the number of inputs, and f, g, and h simply depict three distinct functional forms, we can rewrite Eq. (3) as Eq. (1), and interpret Y as the SNM mass M, so Eq. (1) is alternatively expressed as

$$M = f(X_1, X_2, ..., X_p) \qquad (4).$$

Eq. (4) is intended for bottom-up UQ. But top-down UQ often suggests the need for a measurement error model that allows for both systematic and random errors, such as

$$M = T + S + R \qquad (5),$$

where T is the true SNM mass, R is random error, and S is systematic error that accounts for all unmodeled or incorrectly modelled effects for each item. The systematic error could scale with the true value or not (that is, the error model could be multiplicative or additive or something else), and could vary across items or not, depending on the context [6-11]. If S varies across items, then it is item-specific bias [9,10]. Item-specific bias is nearly always present to some extent, because test items differ to some extent from calibration items. Many NDA examples adjust test items (as do the three examples in Section 3 and example in Section 4) to calibration items using some type of modelling. Model uncertainty must therefore be addressed.

Top-down UQ estimates the R and S error magnitudes, typically quantified by their standard deviations σ_R and σ_S, which are estimated from data sets that have measurements of some or all items from each of two or more assay methods. We use the hat notation to denote estimated quantities; for example, $\hat{\sigma}_R$ and $\hat{\sigma}_S$ denote estimates of σ_R and σ_S, respectively. We use capital letters to denote random variables. The random error term R can include variation in background that cannot be perfectly adjusted for, Poisson counting statistics effects, and other random effects. In principle, the $X_1, X_2, ..., X_p$ could be estimated for each item as part of the assay protocol. However, there would still be modelling error because the function f must be chosen or somehow inferred, possibly using purely empirical data mining applied to calibration data [6,8,11], or physics-based radiation transport codes such as Monte-Carlo-n-particle (MCNP, [12]). Typically, only some of $X_1, X_2, ..., X_p$ will be measured as part of the assay protocol, as we illustrate in the uranium neutron collar case study.

3. The GUM and UQ for NDA

Recall that Eq. (4) is aimed primarily at bottom-up UQ, using either steps in the assay method and uncertainties in $X_1, X_2, ..., X_p$, or using calibration data (see the UNCL case study in Section 4). However, supplements to the GUM describe analysis of variance in the context of top-down UQ using measurement results from multiple laboratories and/or assay methods to measure the same measurand.

The purpose of a measurement is to provide information about the measurand, such as the SNM mass. Both frequentist and Bayesian viewpoints are used in estimating the measurand and in characterizing the estimate's uncertainty. Elster [13] and Willink [14] point out that the GUM invokes both Bayesian and frequentist approaches in a manner that is potentially confusing. To modify the GUM so that a consistent approach is taken for all types of uncertainty, [14] suggests an entirely frequentist approach while others suggest an entirely Bayesian approach. Bich [5] also points out confusion between frequentist and Bayesian terminology and approaches in the GUM, which is one reason it would be useful to revise the GUM. No matter which approach is used, making it clear which quantities are viewed as random and which are viewed as unknown constants will avoid needless confusion. However, the real challenges involve choosing a likelihood for the data, a model to express how the measurand is estimated, and a model to describe the measurement process. These challenges are present in both frequentist and Bayesian approaches.

Ambiguities in the GUM arise for at least three reasons [3,14,16]: (**1**) The GUM divides the treatment of errors into those evaluated by type A evaluation (traditional data-based empirical assessment), and those addressed by type B evaluation (expert opinion, experience with other similar measurements). However, type B evaluations are primarily Bayesian (degree of belief) without explicitly stating so (and need not be), while type A evaluations are primarily frequentist (and need not be). The jargon used in describing type B evaluations implies that the true value *T* has a variance (a Bayesian view based on quantification of our state of knowledge). The jargon used in describing type A evaluations is frequentist, with statements such as $P(X-T>k_1\sigma)=0.05$, with the interpretation that *X* varies randomly around the fitted quantity *T* , where σ is the known measurement error standard deviation. We endorse either view, when clearly explained, but typically write $P(X-T>k_2\hat{\sigma})=0.05$ where the hat notation conveys that the standard deviation is an unknown parameter that must be estimated, so $k_2>k_1$. (**2**) The GUM uses the same symbol *X* for a measurement result and for a true value, which also confuses the frequentist and Bayesian views. (**3**) There is vague use of the term "quantity." And, although the GUM attempted to clarify confusion between "error" and "uncertainty," it did not clearly use the term "error" when measurement error (which has a sign, positive or negative) was meant. Willink [14] aims to resolve these ambiguities by paying attention to notation and jargon, being careful to separate Bayesian from frequentist views, and pointing out a confusion of true values with measurements of true values. Also, the GUM does not explicitly address calibration; however, because calibration is almost never a completely straight-forward application of ordinary regression, we agree with [13] that UQ for calibration deserves attention, as we illustrate with the UNCL example in Section 4.

Elster [13] points out that Eq. (4) is Bayesian because it implies a probability distribution for the SNM mass *M*, and [13] shows that for a particular form of noninformative (large variance) prior probability distribution for *M*, there is Bayesian approach that exactly agrees with that implied by placing a joint probability distribution on the inputs. We point out here that historically, Bayesians have regarded the posterior probability distribution as the central feature of a Bayesian analysis. Frequentists interpret probability as the frequency of occurrence of an event. In metrology (for NDA or more generally), there is opportunity to merge some of the best Bayesian and frequentist practices. Prior probabilities can encode constraints, such as true quantities being nonnegative; then, the actual frequency within which a 95% Bayesian probability interval actually includes the true value can be observed; thus, metrology provides a practical application for the notion of being a "calibrated Bayesian." A calibrated Bayesian borrows Bayesian and frequentist ideas.

3.1 Example application of GUM to NDA

Nearly all assay methods, including all NDA methods rely on calibration. However, NDA methods are sometimes applied to test items that have different physical properties than calibration items. Elster [13] shows why GUM's measurement Eq. (1) is not directly set up for calibration. However, with some creativity, one could map calibration problems to Eq. (1), which we now illustrate using a simple version of the enrichment meter principle (EMP).

Suppose we fit the known enrichment in each of several standards to observed counts in a few energy channels near the 185.7 keV energy as the "peak" region and to the counts in a few energy channels just below and just above the 185.7 keV energy to estimate background, expressed as

$$Y=\beta_1 X_1+\beta_2 X_2+R \qquad (6),$$

where *Y* is the enrichment, X_1 is the observed peak count rate, X_2 is the observed background count rate, and *R* is random error. The calibration data is used to estimate β_1, and β_2. One could constrain the estimates $\hat{\beta}_1$ of β_1 and $\hat{\beta}_2$ of β_2 to be equal in magnitude in the case where the same number of energy channels is used for both the peak and background. That would correspond to assuming a constant (non-sloping) background throughout the peak region, which is sometimes, but not always, appropriate. Therefore, in practice, we do not force the constraint $\hat{\beta}_1=-\hat{\beta}_2$. Also, note that the true enrichment is never known exactly, not even in standards; however, the uncertainty in standards can be accounted for, and for convenience here will be assumed to be negligible.

Because X_1 and X_2 are the measured count rates, they have measurement error. However, as we show numerically in the UNCL case study, there is no need to use the errors in predictors literature [7,17]. There is the need to estimate the 2-by-2 covariance matrix of $(\hat{\beta}_1,\hat{\beta}_2)$, which is best done by simulation unless one can safely assume that the errors in X_1 and X_2 can be neglected in this context. Some will argue that using simulation to estimate cov$(\hat{\beta}_1,\hat{\beta}_2)$ is beyond application of GUM's Eq. (1). However, if we adopt a general interpretation of Eq. (1), allowing type A and/or type B analyses to inform on the probability distribution of the inputs, then we can compute the estimated *Y* using $\hat{Y}_{\text{test}}=\hat{\beta}_1 X_{1,\text{test}}+\hat{\beta}_2 X_{2,\text{test}}$, which can be thought of as being an example of $Y=f(X_1,X_2,X_3,X_4)=f(\hat{\beta}_1,\hat{\beta}_2,X_{1,\text{test}},X_{2,\text{test}})=\hat{\beta}_1 X_{1,\text{test}}+\hat{\beta}_2 X_{2,\text{test}}$, which has a probability distribution that can be inferred from simulation applied to the calibration data. If the count times in training differ from the count times in testing, then modifications are necessary. Here the term "training" refers to the calibration items used for parameter estimation; and the term "testing" refers to calibration and/or test items that are not used for parameter estimation, but that are used for performance estimation. Also, see [13], who points out that there is not a unique functional form analogous to GUM's Eq. (1), $Y=f(X_1,X_2,...,X_p)$. Instead, there is a collection of $(\mathrm{Y},X_{1,\text{train}},X_{2,\text{train}})$ triples from which $(\hat{\beta}_1,\hat{\beta}_2)$ is estimated, using, for example, errors in predictors methods or not. So, even relatively simple calibration applications such as the EMP without complications (next paragraph) are not fully treated by GUMs measurement Eq. (1).

Nevertheless, one could defend regarding $\hat{\beta}_1 X_{1,\text{test}} + \hat{\beta}_2 X_{2,\text{test}}$ as a deterministic function of random quantities, and so it has a probability distribution, paving the way for a Bayesian treatment if desired.

Several departures from calibration items can occur in test items; one common departure from calibration items is that test items could have meaningfully different container thicknesses, which must be measured and then the count rates X_1 and X_2 are adjusted accordingly, using the factor exp(mr$x$), where x is container thickness, mr is the gamma linear absorption coefficient that adjusts for test items and calibration items having different container thicknesses. Reference [2] gives more detail about calibration and analyses of EMP data. If such departures occur, then a model is used to adjust to calibration conditions, which might still be amenable to a GUM-type UQ analysis, depending on the complexity of the model and the methods needed to validate the model for the NDA application.

3.2 Example NDA applications that require extensions to the GUM

This subsection describes two NDA applications (the Cf shuffler, and the distributed source term analysis) that have emerging UQ topics that are not specifically addressed by the GUM, including: item-specific biases, errors in predictors, and quantification of uncertainty in computer models. Many NDA applications are like the extended version of the EMP where model-based adjustments are needed to adjust physical attributes of test items to those in the calibration items.

3.2.1 Cf Shuffler

Shufflers measure fissile masses nondestructively by counting neutrons released as a result of fissions that are induced by successive irradiations from a source consisting of ^{252}Cf neutrons. As the hydrogen density from the non-SNM material increases, the shuffler accuracy can degrade because the detected count rate varies with the SNM positions within the item. In some cases, hardware additions to reduce the average energy of the irradiating neutrons reduce this problem, but increase item self-shielding, leading to other bias sources. Alternate strategies, including imaging, have been pursued, but none have been completely acceptable.

Certified standards exist for only a few material categories (categories are defined on the basis of material type and packaging), while measurements are needed for a wide variety of categories. The standards are used in a calibration step to estimate the pseudo-source strength of the ^{252}Cf neutrons. The procedure is to determine the source strength that minimizes the difference between the measured and MCNP-calculated count rates (CR) for the standards. The calculated count rate is the product of three numbers, $F_1 \times F_2 \times F_3$, where F_1 is a MCNP-based estimate of the expected total number of counts (over all shuffle cycles) per source ^{252}Cf neutron per gram of SNM, F_2 is a known constant that is determined by cycle time parameters such as count times with and without the ^{252}Cf source neutrons, and F_3 is a calibration parameter to be estimated [18]. The equation M = CR/($F_1 \times F_2 \times F_3$) + error can be fit to the measured data for calibration standards, leading to a least squares estimate of F_3. Because of non-negligible error in F_1 due to MCNP-based model uncertainty, this is another "errors in predictors" problem. However, in this case, there is no direct interest in the estimate of F_1, F_2, or F_3. The main goal is to use the estimated $F_1 \times F_2 \times F_3$ to convert CRs on test items to estimates of SNM mass.

If there were no model error (and all the relevant properties of the standards such as density and material form were known exactly), then F_3 would estimate the actual ^{252}Cf neutron source strength. Following calibration on standards to estimate F_3, we convert measured CR on a test item to estimated SNM mass using M = CR/($F_1 \times F_2 \times F_3$) + error. There will be errors in the predictor CR due to having a modest total count time, and there will be errors in the product $F_1 \times F_2 \times F_3$ arising from errors in F1 × F3 due to modelling imperfections. MCNP modelling is also used to model how detection efficiency is impacted by the fissile material positions because different fissile material positions imply different neutron travel distances through the matrix toward the detectors [18].

An unusual aspect of the calibration procedure is the partitioning into measurement categories. From the procedure described above, we recognize that if a new material category is to be assayed, using the F_3 associated with calibration on the standards, then model errors arise from: (1) unmodeled effects that impact the new material category in a different manner than they impact the standards, and/or (2) improperly specified material properties in either the standards or the new category. Either of these effects leads to errors in $F_1 \times F_3$ that could be different for the new category than for the standards and also different than errors in other measurement categories.

Examples of unmodeled or inaccurately modeled effects include some of the following: (1) the Cadmium liners on the detector banks have holes that are not currently modeled; (2) The detector is not technically modeled exactly as built. It would take a great deal of effort to include all the details, for examples there are certainly air gaps in the assembly of polyethlene blocks, yet the blocks are assumed to be one solid mass;(3) The ^{252}Cf neutrons are inside of a small metal capsule, yet this capsule is not included in the model for cost/benefit reasons. (4) There is a motor below the rotating turntable in the floor of the shuffler; it was not put in the model for cost/benefit reasons. This example list is current but subject to change; however, for any implementation there will be unmodeled effects, some of which could be important in UQ, although it is usually assumed that biases due to such modelling imperfections largely cancel out in relative calculations.

We regard the Cf shuffler with computational adjustments ($F_1 \times F_3$ is estimated from imperfect application of MCNP) as having uncertainty that, while in principle, might be amenable to a type B evaluation, regulators will not yet accept the computational adjustments without further study. In short, the "propagate uncertainty in inputs to uncertainty in the output" guidance as implied by GUMs Eq. (1), is useful, but still leaves most of the work to understanding uncertainty in the inputs (which include computational adjustments) to those performing bottom-up UQ. Also, fundamental nuclear data, which is used, for example, by MCNP in the Cf shuffler example, and in many other examples, has uncertainties whose impact depends on how calibration data and models are used.

3.2.2 Distributed Source Term Analysis

The distributed source-term analysis (DSTA) is a measurement technique that has been applied to a variety of safeguards and verification situations where the amount of neutron-producing material contained within a large area needs to be measured in a timely fashion. The technique was originally developed to assay material present in uranium enrichment cascade halls [19] using neutron counting. It has also been applied to the assay of low-activity waste storage areas, and static material storage areas. Current development of the DSTA technique is focused on material accountancy in plutonium glovebox process lines. In these cases, tools are being developed to assist the operator to localize materials to improve cleanout operations.

The DSTA is applied in situations where a neutron assay of a large area is required. In these cases, the sampled area is too large to be placed inside a counter as is done in a traditional neutron assay measurement. Instead, the detector is positioned at a variety of known positions within the sample volume and the neutron count rate is measured at each position. The sample area is then divided into a number of discrete source voxel locations. A room-response matrix is determined using MCNP to estimate the source-to-detector coupling for each voxel-measurement position pair. The MCNP-based estimate of the room response and measurement data are then used to estimate the neutron activity in each source voxel in the assay area.

The DSTA analysis consists of three basic activities: sampling, simulation, and fitting. During sampling, measurement positions are selected throughout the assay area to ensure that the entire sample volume is adequately measured, where adequacy is determined by the particular application. The measured counts for positions 1, 2, …, P, are placed in a vector M_P ([counts/sec]) and the location of each measurement position is recorded for use in the simulation phase.

In the simulation phase, the sample itself (large storage area, process hall, etc.) is modeled using MCNP. The sample can be divided into V discrete source voxels, the activity of which (A_V [neutrons/sec]) can be estimated from the DSTA method, provided the number of measurement positions P > V. The MCNP code is used to determine the source-to-detector coupling between each of the *V* source voxels and each of the *P* measurement positions. The MCNP efficiency results are used to populate a response matrix, R_{VP} ([counts/neutron]). The resulting system of linear equations is used to estimate the neutron production activity of each source voxel in units of neutrons/second.

The DSTA data model is then $M_p \sim \text{Poisson}(t\sum_V R_{VP}A_V)\Big/t,$

and the main goal is to estimate the total activity, $\sum_V A_V$ (or total SNM mass). This is another errors-in-predictors problem, where the predictors R_{VP} are estimated using MCNP and the estimates are partially validated using real (and corresponding MCNP-modeled) Cf sources at known source locations and recording their measured source strength at known detector locations. Because the error structure in the R_{VP} matrix is currently not well known, [20] could only perform a "what if" sensitivity study, simply evaluating the error in the estimated total activity, $\sum_V A_V$ under various assumptions about the error structure in the MCNP-based estimate of R_{VP},and using various options for dealing with errors in predictors. None of the errors in predictors literature deals with the types of error structure that are possibly present in $R_{VP,}$ such as having both random and systematic components. As in the EMP example, one could map the data model $M_p \sim \text{Poisson}(t\sum_V R_{VP}A_V)\Big/t$ to one that expresses the estimate for total activity, $\sum_V A_V$ as a function of several inputs, including MCNP-based estimates of $R_{VP.}$ However, this again still leaves most of the work to understanding uncertainty in the inputs (which include MCNP-based estimates of R_{VP}) to those performing bottom-up UQ.

4. UNCL case study

The UNCL uses an active neutron source (AmLi) to induce fission in the ^{235}U in fresh fuel assemblies [21-23]. Figure 1 is a simple overhead view produced by MCNP [12, 21-23]. Neutron coincidence counting is used to measure the "reals", i.e. neutron coincident rate *R* attributable to fission events, which can then be used to determine the linear density of ^{235}U in a fuel assembly (g-^{235}U/cm) from calibration parameters, a_1 and a_2. The equation used to convert the measured *R* to *Y* (gms ^{235}U per cm) is

$$Y = \frac{kX}{a_1 - a_2 kX} \qquad (7),$$

where a_1 and a_2 are calibration parameters, and $k = k_0k_1k_2k_3k_4k_5$ is a product of correction factors that adjust *R* (*R* = *X* in Eq. (7)) to item-, detector-, and source-specific conditions in the calibration [21-23]. Therefore, Eq. (7) is a special case of GUM's Eq. (1), where the two calibration parameters a_1 and a_2 and the 6 correction factors k_0,k_1, k_2, k_3, k_4,and k_5 are among the X's in Eq. (1). We caution readers that GUM does not fully treat multi-parameter calibration uncertainties, so there are open issues in applying GUM's Eq. (1) even to this relatively straightforward calibration problem. Also, there is much current research on options to improve the UNCL method for new types of fuels, which will be reported elsewhere. Nevertheless, it provides a practical basis to support discussion of the current practice for NDA and to describe a roadmap for more comprehensive UQ for NDA.

4.1 Description of UNCL Calibration and the 6 Correction Factors k_0, k_1, k_2, k_3, k_4, and k_5

Menlove et al. [22] introduced correction factors to adjust the measured reals count rate to the corresponding reals count rateobserved in the calibration condition for a particular a_1, a_2 coefficient pair. Coefficient-pairs were defined for standard PWR and BWR fuel types by [16]. Since that original reporting coefficient pairs have been determined for WWER-440 and WWER-1000 fuel types [21-23].

The term k_0 accounts for uncertainty in the true Am/Li source strength (approximately historically 3.7% relative error standard deviation (RSD) if using recent IAEA estimates). The term k_1 accounts for uncertainty due to electronic drift (considered negligible with modern electronics, so k_1=1). The term k_2 accounts for uncertainty due to differences in detector efficiencies (approximately 1.5% RSD). The term k_3 accounts for the effects of burnable poison (burnable poison absorbs neutrons). The term k_4 accounts for differences in the total uranium loading (U-total/cm) between the calibration case and the measurement case. The term k_5 accounts for all other effects (eg spacers, bagged assemblies).

The k-factors were introduced to allow for using the same a_1 and a_2 values over a wide range of measurement cases and different UNCL detector systems. In the present consideration the calibration factors, a_1 and a_2, and the k-factors help to identify error sources in the UNCL measurement and calibration.

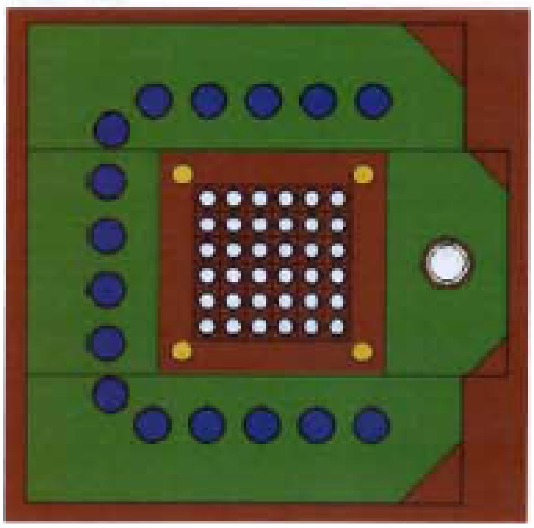


Figure 1: Simple overhead view of an UNCL, produced from MCNP. The 6x6 fuel pins of the assembly are in the center of the detector cavity in white. The 16 blue circles forming a "collar" around the sample are the ^{3}He neutron proportional counters. The green area represents the polyethylene moderator surrounding the detectors; the red areas are air. The source is the large white circle towards the right.

4.2 Example analyses

We reanalysed 9 pairs of (R, ^{235}U) from Table VII for PWR from [22], fitting Eq. (7) with approximately 2% RSD. Figure 2 plots the 9 (R, ^{235}U) pairs. We then applied a single noise factor and k = k_0 k_1 k_2 k_3 k_4 k_5 to introduce noise due to departure from calibration conditions as described in Section 4.1.

Figure 3 gives example RSD values for the 9 (R,^{235}U) pairs in 10^5 simulations in R [24]. In each simulation, 6 of the 9 (R,^{235}U) pairs were randomly selected to calibrate, and the other 3 (R,^{235}U) pairs were used to test. Dividing into training and testing helps to account for model uncertainty in Eq. (7). Varying amount of random error in k was applied, ranging from approximately 1 to 5% RSD, which represents the aggregate effect of errors in each of k_0-k_5. The plot in Figure 3 assumed that the same RSD values in k were present in the 6 training pairs as in the 3 testing pairs. If there are different error magnitudes in testing than in training, then bias can be introduced in the estimated ^{235}U [3,7]. Also, if there is an adjustment for errors in predictors [7,17], then the RSD is higher (option 2 in Figure 3) compared to not adjusting for errors in predictor (option 1 in Figure 3). And, there is a very large bias component contributing to the large RSD in the option 2 results in Figure 3. Interestingly, there is sometimes a large bias being observed in top-down evaluations of the UNCL [3]. The adjustment for errors in predictors is to choose values of $x_{i,true}$ and a_1 , a_2 to minimize

$$RSS_1 = \sum_{i=1}^{n_{train}} \frac{(x_i - \hat{x}_i)^2}{\sigma_{x_i}^2} + \frac{(y_i - \hat{y}_i)^2}{\sigma_{y_i}^2} \quad ,$$

where $\hat{x}_i$ is the estimate of $x_{i,true}$ is the first term, and $Y = \frac{kX}{a_1 - a_2 kX}$ is used to calculate $\hat{y}_i$ in the second term (using $\hat{x}_i$, the estimate of $x_{i,true}$ in the expression $\hat{y}_i = \frac{kx_i}{a_1 - a_2 k\hat{x}_i}$). The weights $\sigma_{x_i}^2$ and $\sigma_{y_i}^2$ are assumed here to be known; we used a range of possible values (approximately 1 to 5% RSD) for $\sigma_{x_i}^2$ (which includes the effect of errors in R and in k) and we used the residual variance from the fit of ^{235}U to R (with k set equal to 1) using all 9 (R, ^{235}U) pairs for $\sigma_{y_i}^2$. If there is no adjustment for errors in predictors, then a_1 , a_2 are chosen to minimize $RSS_2 = \sum_{i=1}^{n_{train}} \frac{(y_i - \hat{y}_i)^2}{\sigma_{y_i}^2}$, which is the appropriate and familiar criterion if the goal is to predict y.

GUM's key measurement equation (our Eq. (1) and (4)), $M = f(X_1, X_2, ..., X_p)$ could be modified to allow for both "computational calibration" in which one uses modelling to adjust test items to calibration items (via the *k* factor in the UNCL example) and errors in predictors. However, one would need to allow for bias in the adjustment to calibration items by having different probability distributions for some of the inputs $X_1, X_2, ..., X_p$ in training and testing. This moves the analysis toward Monte Carlo simulation assessments, without concern for whether the probability distribution for the item SNM mass can be expressed as a known function, $M = f(X_1, X_2, ..., X_p)$. Because we do not yet have a defensible estimate of the probability distribution of the *k* factor, this example is for illustration only, not for a defensible bottom-up UQ for the UNCL. Also, choices in how to perform the calibration (with or without adjusting for errors in predictors for example) are best assessed using simulation, as we did in Figure 3.

5. Discussion and Summary

This article has illustrated several challenges in UQ for NDA (EMP, Cf shuffler, DSTA, and UNCL). As the need for better UQ for NDA is becoming recognized, the GUM [4] is being revised [5]. It is possible that the NDA community will need a modified GUM, or that NDA UQ needs can influence the in-progress GUM revision. For example, the UNCL case study illustrates that there is a need for attention to errors in predictors in the GUM supplement that deals with calibration. The UNCL case study is also an example of "computational calibration," in which one uses modelling to adjust test items to calibration items. Other examples of computational calibration include the Cf shuffler [18], and possibly, in new applications of NDA to spent fuel assay [8]. In the case of spent fuel assay, it is currently unclear to what extent MCNP modelling will be used as part of the assay procedure once working standards become available.

6. Acknowledgements

The authors acknowledge the IAEA, and US Nuclear Nonproliferation Agency NA-22.

7. References

[1] McElroy, R., Croft, S., Nakazawa, D., Kirkpatrick, J., Venkataramin, R., Burr, T., The need and opportunity for improved uncertainty quantification and reporting for nondestructive assay, Proceedings Institute of Nuclear Materials Management, 2012.

[2] Burr, T., Trellue, H., Tobin, S., Croft, S., Uncertainty quantification challenges in nondestructive assay of nuclear material, Conference on Data Analysis, Santa Fe, NM, 2014.

[3] Bonner, E., Burr, T., Guzzardo, T., Norman, C., Zhao, K., Beddingfield, D., Geist, W., Laughter, M., Lee, T., Improving the effectiveness of safeguards through comprehensive uncertainty quantification, to appear, Journal of Nuclear Materials Management, 2016.

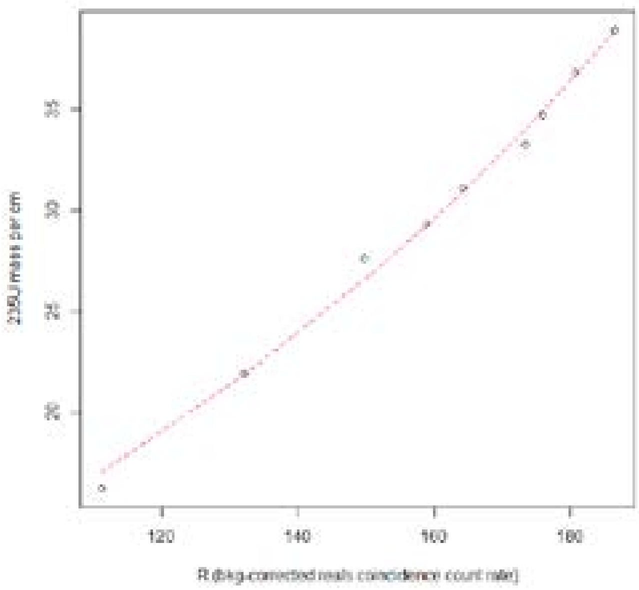


Figure 2: The ^{235}U mass per cm (linear density) versus the background-corrected reals coincidence rate R. The fit to Eq. (7) from the data is also shown. The 9 R values are 111.1, 132.0, 149.7, 158.8, 164.1, 173.4, 176.0, 180.8, 186.5 [/s]. The corresponding 9 ^{235}U values are 16.20, 21.89, 27.59, 29.37, 31.15, 33.28, 34.71, 36.84, 38.98 [g/cm].

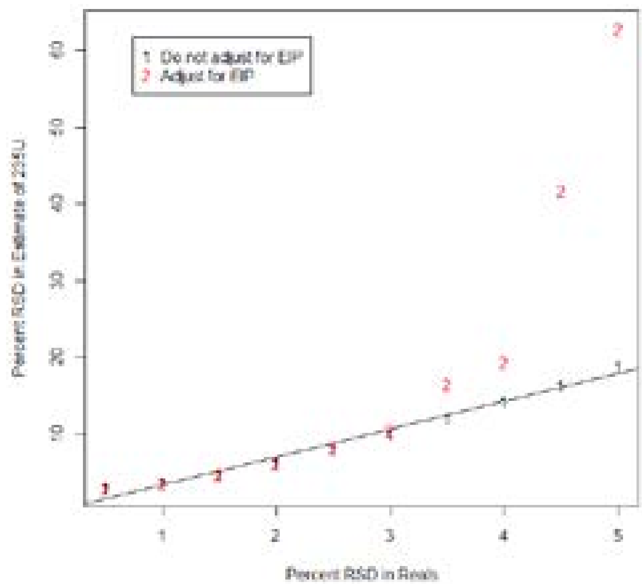


Figure 3: The RSD in UNCL prediction versus RSD in *k*R on the basis of 10^5 simulations. The data are the 9 data pairs from [22]. A large bias component is contributing to the large RSD in the option 2 results, due to adjusting for EIP by using the RSS_1 criterion rather than the RSS_2 criterion to estimate a_1 , a_2.

[4] Guide to the expression of uncertainty in measurement, JCGM 100: 2008, www.bipm.org

[5] Bich, W., Revision of the guide to the expression of uncertainty in measurement. Why and How, Metrologia 51, S155-S158, 2014.

[6] Burr, T., Pickrell, M., Rinard, P., Wenz, T., Data mining: applications to nondestructive assay data, Journal of Nuclear Materials Management 27(2), 40-47, 1999.

[7] Burr,T., Knepper, P., A study of the effect of measurement error in predictor variables in nondestructive assay, Applied Radiation and Isotopes 53 (4-5), 547-555, 2000.

[8] Burr, T., Trellue, H., Tobin, S., Favalli, A., Dowell, J., Henzl, V., Mozin, V., Integrated nondestructive assay systems to estimate plutonium in spent fuel assemblies Nuclear Science and Engineering 179(3), 321-332, 2015.

[9] Burr, T., Sampson, T, Vo, D., Statistical evaluation of FRAM γ-ray isotopic analysis data, Applied Radiation and Isotopes 62, 931-940, 2005.

[10] Burr, T., Hemphill, G., Multi-component radiation measurement error models, Applied Radiation and Isotopes 64(3), 379-385, 2006.

[11] Burr, T., Dowell, J., Trellue, T., Tobin, S., Measuring the effects of data mining on inference, Encyclopedia of Information Science and Technology, Third Edition, 2015.

[12] Monte Carlo N- Particle code, see mcnp.lanl.gov at Los Alamos National Laboratory.

[13] Elster, C., Bayesian Uncertainty analysis compared to the application of the GUM and its supplements, Metrologia 51, S159-S166, 2014.

[14] Willink, R., Measurement uncertainty and probability, Cambridge University Press, Cambridge, 2013.

[15] Standard test method for measurement of 235U fraction using the enrichment meter principle, C 1514-08, ASTM.

[16] Elster, C., Toman, Bayesian uncertainty analysis for a regression model versus application of GUM supplement 1 to the least-squares estimate, Metrologia 48, 233-240, 2011.

[17] Fuller, W., (1987). Measurement Error Models, Wiley, New York.

[18] Tobin, S., Swinhoe, M., and MacArthur, D., Monte Carlo Study of Replacing Cf in a Shuffler with a Neutron Generator, LA-UR-04-4354. Institute of Nuclear Materials Management Annual Proceedings, 2004.

[19] Beddingfield, D.H., and Menlove, H.O., Distributed source term analysis, a new approach to nuclear material inventory verification, Nuclear Instruments and Methods-A, 485, 797-804, 2002.

[20] Beddingfield, D., Burr, T., Longo, C., Error evaluation in distributed source-term analysis, LAUR10-01947, 2010.

[21] Favalli, A., Croft, S., Swinhoe, M., Perturbation and burnable poison rod corrections for bwr uranium neutron collar, Proceedings ESARDA, 2011.

[22] Menlove, H., et al, Neutron collar calibration and evaluation for assay of LWR fuel assemblies containing burnable neutron absorbers, Los Alamos National Laboratory Report, LA-11965-MS (also ISPO-323), 1990.

[23] Peerani, P., Computational calibration of UNCL neutron collars for fresh fuel elements, Technical Note I.04.142, European Commission Joint Research Center, 2004.

[24] R Core Team (2012). R: A language and environment for statistical computing. R Foundation for Statistical Computing, Vienna, Austria. ISBN 3-900051-07-0, www.R-project.org.

Mobile 3D Laser Scanning for Nuclear Safeguards

E. Wolfart, S. Ceriani, D. Puig, C. Sanchez, P. Taddei, V. Sequeira
Joint Research Centre, European Commission, Ispra, Italy
M. Murtezi, P. Turzak, A. Zein
DG ENER, European Commission, Luxembourg
L. Enkhjin, M. Ingegneri, S. Rocchi, Y. Yudin
International Atomic Energy Agency, Vienna, Austria

Abstract:

3D laser scanning is an established verification technology in nuclear safeguards, applied inter alia for Design Information/Basic Technical Characteristics Verification (DIV/BTC) and change monitoring in nuclear facilities. Current systems are based on high-accuracy, high-resolution 3D laser scanners which require one minute or more to acquire a single scan. Therefore, the scanners need to be immobile during data acquisition. In order to cover the complete scene, several scans are acquired in a so-called 'stop-and-go' mode, which are then registered into a single coordinate frame in an offline post-processing phase.

Recently, new 3D laser scanners with a significantly increased acquisition speed have emerged. They acquire 3D scans at a frame rate of 10Hz and more - at the cost of reduced accuracy and resolution – and thus enable the scanner to be mobile during acquisition, i.e. the data can be acquired while walking or driving. Mobile laser scanning can significantly increase the efficiency of existing safeguards applications for 3D laser scanning, i.e. DIV/BTC and change monitoring.

Furthermore, by registering each scan with a reference model (which can either be generated a priori or while scanning), it is possible to compute the current position and track the movement of the scanner. Hence, mobile laser scanning with real-time data processing provides indoor positioning capability to nuclear inspectors during their field work. It enables all observations and measurements to be connected with their respective location and time stamps and to retrieve location-based information as required.

The paper presents the Mobile Laser Scanning Platform (MLSP) developed at the JRC, which consists of a commercial mobile scanner, the processing unit and the proprietary software for real-time processing and visualization. The system will be illustrated using two test cases: a DIV/BTC scenario for the future Finnish underground repository (ONKALO) and indoor localization.

Keywords: 3D scanning, Design Information Verification, BTC Verification, change analysis, indoor localization, nuclear safeguards

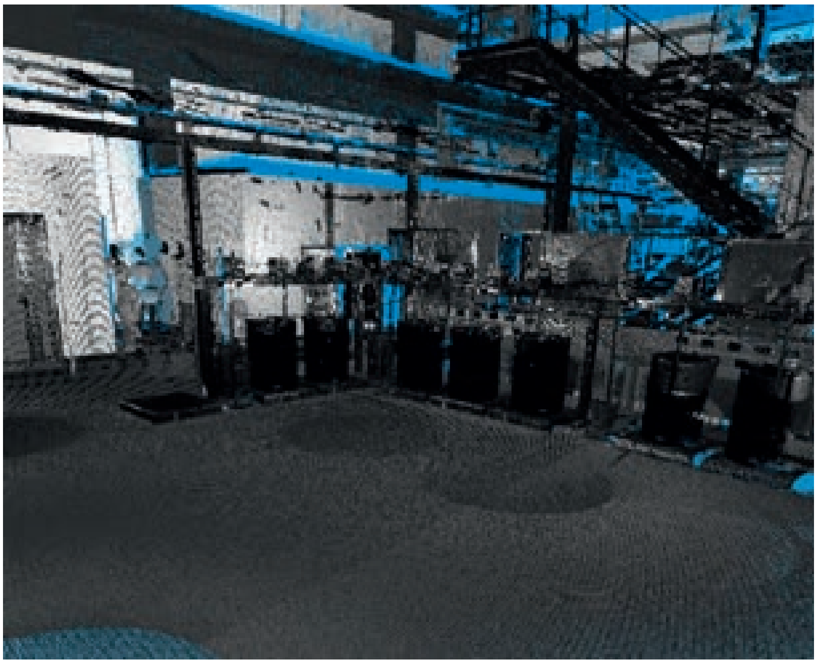
1. Introduction

3D laser scanning is an established verification technology in nuclear safeguards and is approved by IAEA and EC for safeguards use. It has been applied inter alia for Design Information/Basic Technical Characteristics Verification (DIV/BTC) in several nuclear facilities throughout the world, both by IAEA and Euratom inspectors.[1], [2]

Current systems are based on high-accuracy, high-resolution 3D laser scanners which require one minute or more to acquire a single scan. Therefore, the scanners need to be immobile during data acquisition. In order to completely cover a given area of interest, several scans are acquired in a so-called 'stop-and-go' mode, which are then registered into a single coordinate frame in an offline post-processing phase. The resulting 3D model is used to verify the correctness and completeness of the design drawings provided by the operator and it is stored as a reference for subsequent visits. On return, the inspector re-scans the area of interest and the data is analysed to verify that no undeclared modifications to the facility have occurred. Figure 1 illustrates the use of 3D laser scanning for detecting changes in a facility. The change map is calculated from the distances between 3D measurements acquired before and after the scene was changed.

Although 3D laser scanning provides detailed and accurate as-built information and change analysis, data acquisition and processing using stop-and-go scanning can be a considerable effort depending on the size and complexity of the facility. Recently, new 3D laser scanners with a drastically increased frame rate have emerged. They acquire 3D scans at 10Hz and more - at the cost of reduced accuracy and resolution – and therefore allow that the scanner is moved during acquisition, i.e. the data can be acquired while walking or driving.

JRC has developed a portable Mobile Laser Scanning Platform (MLSP), intended for real-time change monitoring inside nuclear facilities, in particular geological final repositories. It is also applicable for indoor localization which allows nuclear inspectors to associate all measurements and observations made during an inspection with the corresponding location inside the nuclear facility and thus facilitate subsequent analysis and future inspections.



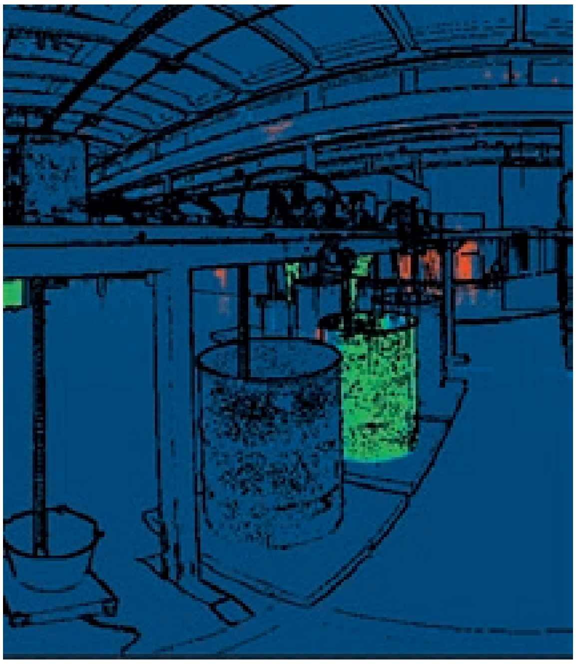


Figure 1: *Left*: snapshot of a 3D model of a (non-nuclear) facility; *Right*: change map generated by comparing the 3D model acquired before and after modifying the scene. Blue pixels correspond to unchanged objects; red pixels correspond to objects that moved closer to the scan position or were inserted to the scene; green pixels correspond to objects that were moved away from the scan position or were removed.

Section 2 describes the main MLSP components and section 3 outlines the core algorithms running on the system. Section 4 illustrates the applications of mobile laser scanning in nuclear safeguards; section 5 outlines future activities and section 6 draws conclusion.

2. Mobile Laser Scanning Platform

The Mobile Laser Scanning Platform (MLSP) is a portable sensor and processing system developed at the JRC. It is based on a mobile 3D laser scanner and provides real-time mapping, localization and change analysis in indoor, GPS-denied environments.

The main hardware components of the MLSP are (see Figure 2):

- A real-time laser range scanner. The current MLSP implementation uses the commercially available Velodyne HDL-32E, which acquires up to 15 frames per second.
- A processing unit which analyses the scanner data in real-time and generates a 3D map/model, localization information and change information.
- A tablet computer to control the system and view the processing results.

The hardware components can be mounted on different carrier systems according to the application need. Figure 3 shows the backpack-mounted MLSP (left) and the car-mounted MLSP (centre, right).

The analysis software which runs on the portable processing unit is the core of the MLSP system. It runs fully automatically in real-time and therefore needs to be highly efficient, reliable and accurate. Additionally, it is able to manage very large data sets, i.e. it is able to handle facilities and buildings that cover several thousands of square meters.

Processing results (tracks and change maps) are transferred to the portable device for visualization and interaction with the user.

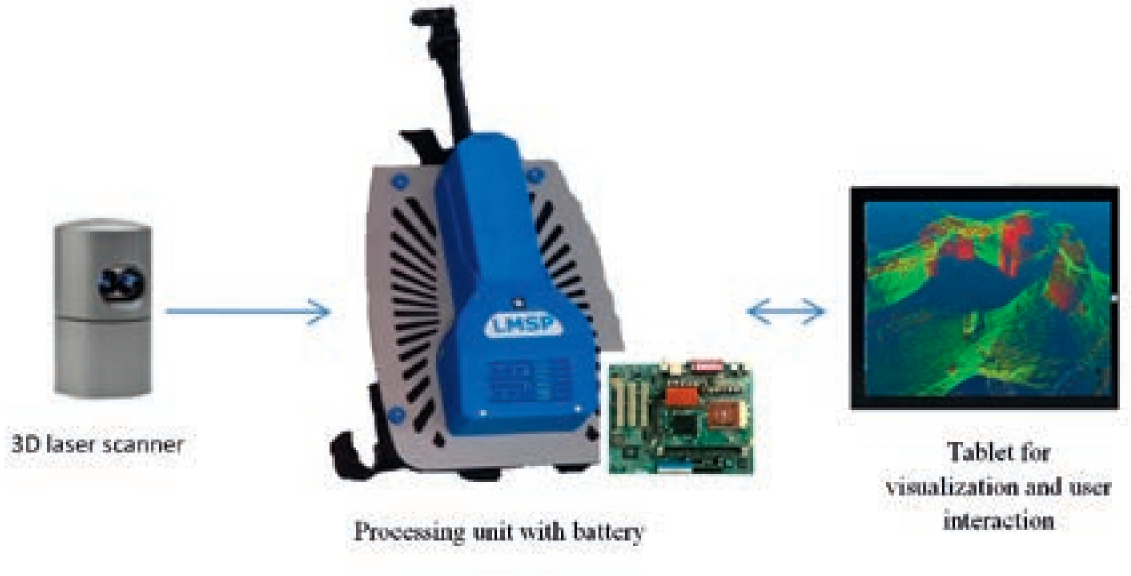


Figure 2: Hardware components of MLSP system: 3D laser scanner, processing unit and tablet.

Figure 3: *Left*: backpack-mounted MLSP for scan-while-walk acquisition. *Centre, Right*: Car-mounted MLSP. The processing unit is situated inside the car.

3. Data Processing

The core of MLSP's data processing is the self-localisation within known environments, i.e. environments for which a 3D reference model has been acquired a-priori. The reference model is typically acquired with the stop-and-go scanning as described above. However, it can also be acquired with mobile scanning if the global accuracy is not essential (see section 4. 2). Since MLSP was developed for indoor use it relies solely on 3D measurements, in particular it does not require any GPS information. The process is divided in two components, which are briefly described in the remainder of this section: *Pose Recognition* and *Pose Tracking*. For a detailed description see [3].

3.1 Pose Recognition

Pose recognition estimates the current user pose (position and orientation) within a given search space (i.e. the space covered by 3D reference model) when no prior knowledge of the current position is available, e.g. after system start-up.

Pose recognition runs in real-time and must be scalable to large environments. Therefore, a pre-processing stage is introduced that (1) reduces the search space of possible poses and (2) transforms the 3D reference model to a compact search tree that enables efficient searching:

- *Search Space Reduction*. Since we focus on ground motion (backpack or vehicle mounted sensor), the MLSP sensor is expected to be in a narrow space parallel to the *navigable floor* and we can reduce the search to the poses within this *navigable space*. Therefore, the navigable floor is computed using on a flooding-algorithm which detects the floor surface based on the surface normals (which are expected to be predominately vertical) and a ''reachable" condition (e.g. a table surface would not be detected as floor because it is not considered to be reachable from the floor surface). We also introduce physical constraints related to the specific mode of system operation (e.g. vertical and angular limits on the possible sensor pose).

- *Search Tree*. In order to enable efficient searching, we transform the 3D reference model into a compact descriptor space as follows: i) we randomly generate a set of poses in the known effective navigable space (computed as described above); ii) for each pose, we synthesize a depth image and extract a compact *descriptor* from the generated depth image: we split the range image in *k* regular bins. For each bin, we estimate a median range value which is stacked to form a single *k*-dimensional descriptor (see Figure 4); iii) we build a binary tree which partitions the *k*-dimensional descriptor space (k-d tree), which is populated with the generated descriptors and the corresponding poses.

Figure 4: Example of a synthesized depth image and the derived compact descriptor. The depth image is divided in twelve bins and the median depth value is calculated for each bin. The descriptor is then stored as a 12-dimensional vector (k=12).

At run-time, the system creates a compact descriptor for each depth image (i.e. 3D scan) in the same way as it is done when constructing the search tree. In order to resolve ambiguities, the pose recognition is based on a temporal series of descriptors and can be divided in an *initialization* and *update* phase as follows:

- During the *initialization, i.e.* when the pose recognition is started and no a priori information is available, the algorithm searches the descriptor space for descriptor/pose pairs matching the current descriptor thus returning a set possible poses of the sensor. Since the descriptors are highly compacted and several positions might have similar geometries (e.g. offices of identical dimensions or positions in long corridors), we typically obtain several ambiguous candidate poses for the initial query.
- The algorithm reduces the set of candidate poses during the *update* phase: as the user navigates within the environment, the algorithm estimates the movement (using the odometer described below) and re-evaluates the likelihood for each initial candidate pose based on the query results at the new pose. The update stage is iterated until the candidate poses converge to a single location and the algorithm is able to disambiguate the current pose. At this point we consider the problem solved and the pose tracking component is started. Figure 5 shows an example of candidate poses after initialisation and after convergence.

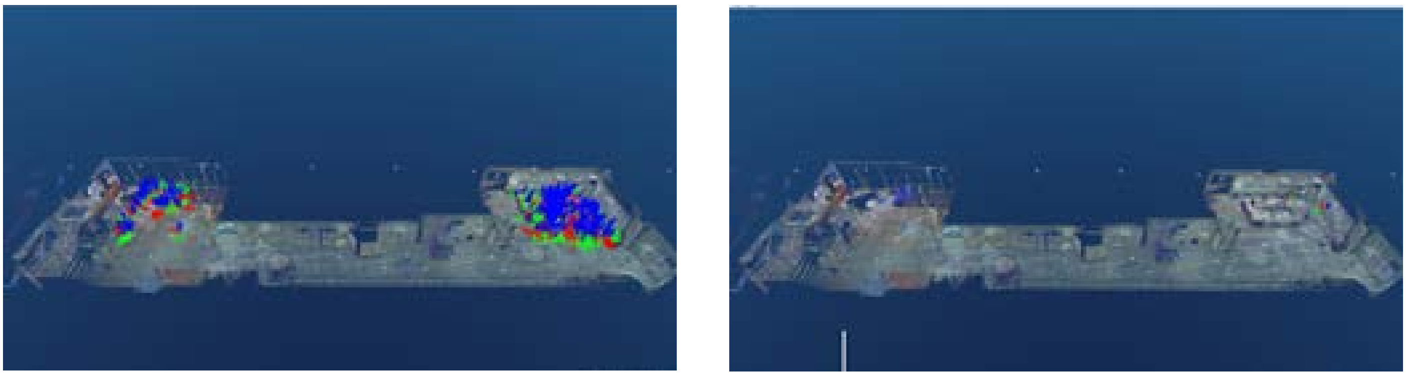


Figure 5: *Left*: Snapshot of pose recognition after initialization. The image shows several ambiguous candidate poses in the two areas that have similar geometries. *Right*: Snapshot of the pose recognition after convergence. The image shows that the candidate poses converge to the correct location (shown as the coordinate frame on the right side) as the user explores the environment for several meters.

3.2 Pose Tracking

Pose tracking starts after the sensor pose has been identified by the pose recognition component. Since the sensor only moves small distances between two scans (i.e. within 0.1 sec), we have a good estimate of the pose of each new scan and it is possible to register the scan using the well-known Iterative Closest Point (ICP) algorithm [4]. Given an initial estimate of a scan pose, the basic ICP algorithm registers a scan with a 3D reference model as follows:

1. Select a sub-set of points from the new scan (control points).
2. For each control point, find the nearest neighbour in the 3D reference model (corresponding points).
3. Compute the transformation that minimises the distance between the control and corresponding points.
4. Update the scan pose using the computed transformation.
5. Iterate steps 1 to 4 until the pose of the scan converges.

The ICP registration accurately estimates the current scan pose (see Figure 6) and therefore the movement since the previous scan, which in turn allows estimating the pose of the next scan. In this way, the ICP is repeatedly applied to each new scan to track the sensor pose as the user moves through the environment.

However, it is challenging to carry out the ICP registration in real-time. The most time consuming step is the nearest neighbour search that has to be carried out for each control point in each iteration. The remainder of this section describes the ICP extensions that were developed to allow real-time ICP processing, namely (1) the transformation of the 3D reference model into a data structure specifically designed for nearest neighbour searches, (2) a point selection and outlier removal strategy to ensure fast and accurate convergence.

3.2.1 Nearest Neighbour Search

In a pre-processing step, two different lists are computed from the 3D reference model: a compact list of points (together with their normals) and a dense grid of voxels. Each voxel can be either *full*, *empty* or *near*. *Full* voxels store an index to an associated point which is computed as the mean of the points inside the voxel. *Empty* cells store a null reference and *near* cells store an index to the nearest plane (see Figure 7, left image).

With this data structure, all nearest neighbour searches are pre-computed offline and stored inside the dense grid. At run time, given a query point in world coordinates, we estimate the nearest neighbour in the map by calculating the voxel that contains it. Then, if the cell state is *full* or *near*, we return the associated point. Otherwise, we notify that there are no neighbours.

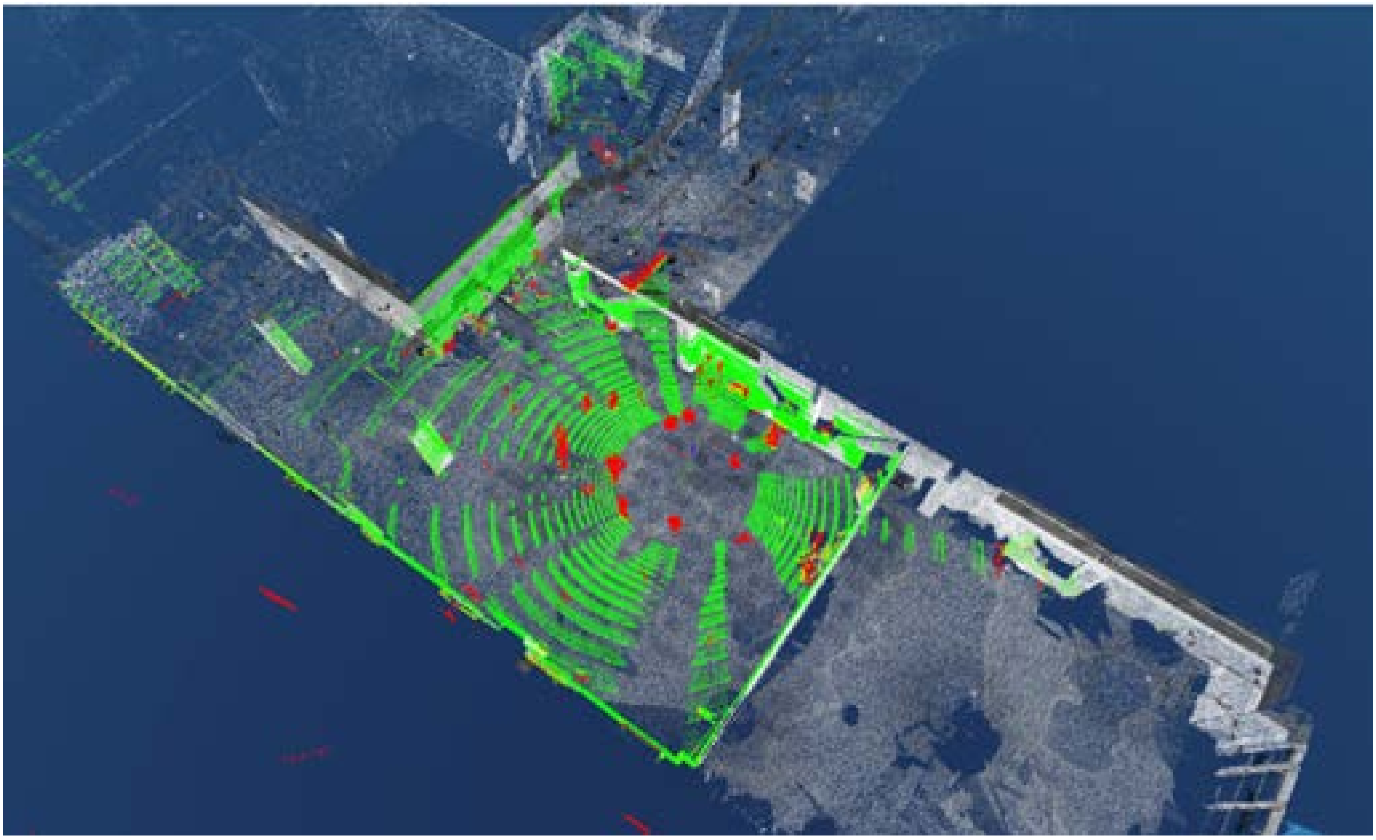


Figure 6: *The green and red points are a scan acquired with mobile scanner; the grey points are the 3D reference model. The current sensor pose is determined by registering the scan with the reference model. The trajectory is obtained by applying the registration repeatedly as the sensor moves through the environment.*

3.2.2 Point Selection and Outlier Removal

For each control point, the nearest neighbour search returns the corresponding point in the reference model. However, outliers (mismatches between control and corresponding points which are, for example, due to objects that have been added or removed after the reference model was acquired) might introduce an error in the computed transformation. In order to allow an accurate and fast convergence of the ICP algorithm, outliers need to be detected and corresponding points need to be selected to properly represent the environment. Therefore, we modify the basic ICP as follows:

- In order to minimise the number of required control points, the selected points should represent all surface directions in 3D space. For example, if we would select only control points from the floor, the registration would not be able to properly lock the current scan in the direction of the walls. Therefore, we create three bins for the principle normal directions of the scan points and the points are classified according to their normals.
- Whereas the basic ICP uses a single set of (typically several hundred) control points, we select many sets, each containing only a small number of points (at minimum 3 points per set are required). The control points are selected from the bins that were pre-calculated as described above.
- For each set, we compute the transformation that minimises distance between control and corresponding points. All computed transformations are distributed around a well-defined central position. However, transformations computed from point sets containing outliers significantly differ from the central position.
- We compute the normal distribution of the transformations and remove the outlier transformations based on their distance to the mean value. This step is iteratively repeated until no transformations are discarded, or a minimum number of transformations is reached.

Figure 7 (right image) illustrates the process of the outlier removal. Notice how all independently computed transformations are distributed around a well-defined central position. Also notice that, after each iteration of outlier removal, the distributions quickly converge to the final estimated transformation, when considering all the correspondences marked as inliers.

3.3 Odometer Integration

The odometer component tracks the movement based only on the sensor data, i.e. without the need of a pre-existing 3D reference model. It works on the same principle as the pose tracking described above, but instead of using the 3D reference model for the ICP registration, the current scan is registered with the previous scan thus generating an estimate of the local movement. Since no global reference model is available, the odometer accumulates drift over time. The odometer is used in two situations:

The pose recognition component uses the odometer to estimate the local movement during the update phase (see above).

If the user leaves the 3D reference model during the pose tracking (e.g. he might enter a room that has not been scanned during the acquisition of the reference model), the track is estimated using the odometer output. The pose tracking automatically reverts to the 3D reference data when the user re-enters the reference model (see Figure 8).

4. Applications of Mobile Laser Scanning in Nuclear Safeguards

MLSP provides indoor localisation with unique accuracy and robustness under the condition that a 3D model of the environment is available. Additionally, it generates a new 3D model of the environment, which can be used for detecting changes with respect to the reference model in real-time:

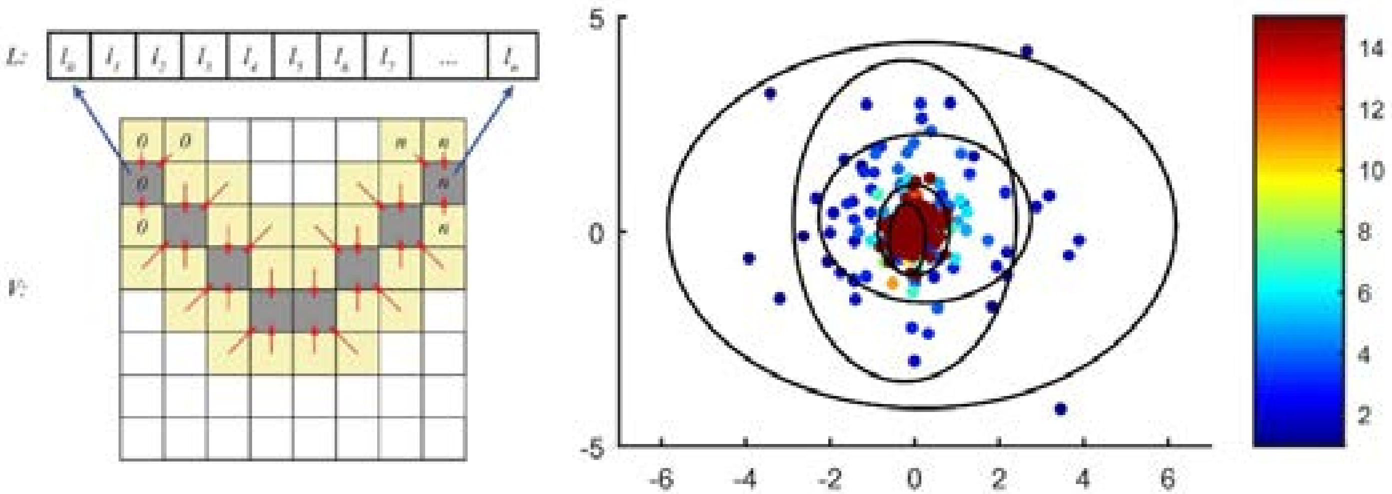


Figure 7: *Left:* Dense voxel structure where, for the sake of clarity, only the closest near voxels are shown. Full cells are displayed as grey boxes. Near cells are represented by yellow boxes with a red line connecting. *Right:* Outlier removal. Axes represent the main dominant dimensions of the detected transformations. Each point represents a candidate transformation coloured according to the iteration in which they have been marked as outliers (some outlier transformations too far from the centre have been omitted). Dark red points represent transformations marked as inliers. The ellipses represent the normal estimations at specific subsequent iterations.

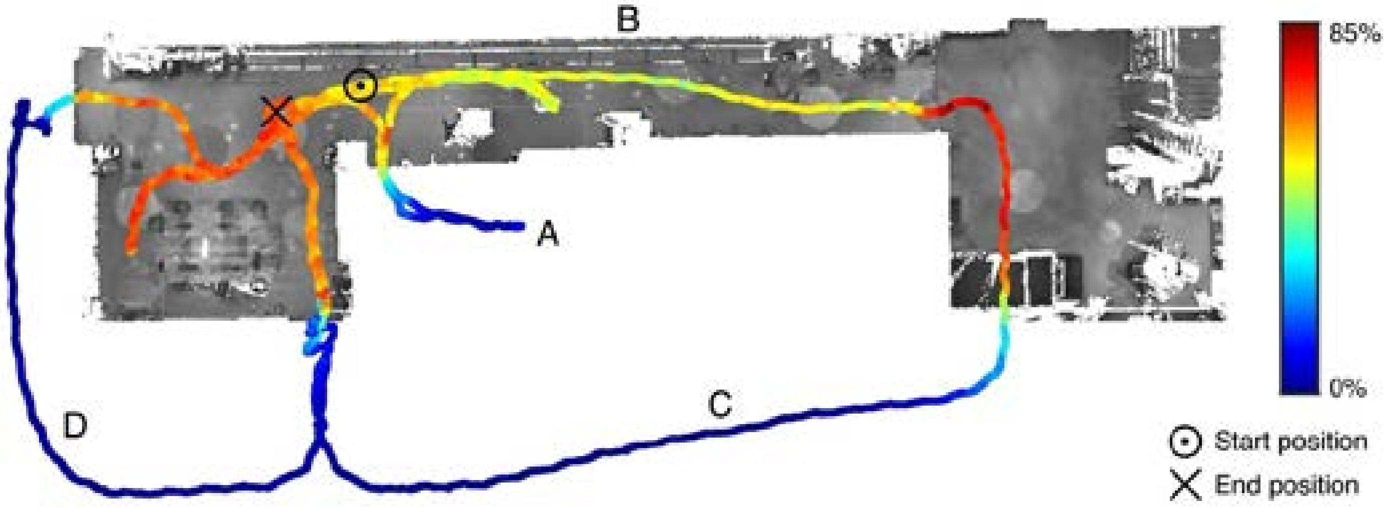


Figure 8: Results of the odometer integration during a sample walk-through inside a building where the user moves to a non-scanned room (A) without losing track of the position. Then, the user performs two loops outside the building (C and D). The trajectory is shaded according to the percentage of points used from the 3D reference model. The rest is taken from the odometer map.

since the 3D scanner is accurately located at any time, an updated 3D model can be generated by simply merging the acquired 3D scans.

Therefore, MLSP is suitable for a series of applications, such as facility management and construction monitoring. In the field of nuclear safeguards, it can be used for position authentication and change monitoring during DIV/BTC verifications and as an enabling technology for location-based applications, for example during Complementary Access inspections. More details are provided below.

4.1 DIV/BTC Verification

Stop-and-go laser scanning is an established verification technology in nuclear safeguards. It allows generating as-built 3D models of nuclear facilities with millimetre accuracy, which then can be compared i) to the design information provided by the operator for DIV/BTC verification or ii) to previously acquired 3D model for change monitoring.

However, stop-and-go data acquisition and the required off-line processing can be a considerable effort in large and complex facilities. In cases where millimetre accuracy is not required, mobile laser scanning is a complementary technology that significantly decreases acquisition effort and provides change information in real time.

The concept is illustrated using the DIV/BTC verification of the future Finnish underground repository in ONKALO. In November 2014, IAEA and DG ENER carried out a DIV/BTC verification at ONKALO using stop-and-go laser scanning. During one week, four teams of inspectors acquired over 900 scans covering more than 6km in total. In parallel, an additional team processed the data to generate an as-built 3D model. At the end, the drawings provided by the operator were verified by comparing them to the 3D model. Figure 9 illustrates the data acquisition and analysis carried out for the 2014 DIV at ONKALO.

The next DIV at ONKALO is scheduled for late 2015. Although some excavations might take place in 2015, the major part of the repository will be the same as in 2014. It is planned to combine stop-and-go and mobile laser scanning as follows:

- The inspector visits the part of the tunnel that already existed in 2014 using the MLSP. The 3D model acquired in 2014 will be used as reference, which will allow to i) always have accurate knowledge of the current location and ii) have real-time information on possible changes with respect to 2014. Figure 10 illustrates the information provided to the inspector in real-time.
- If the inspector identifies any significant changes or makes other relevant observations, he can add notes and comments, which will be location-tagged based on the MLSP information and stored for later reporting, analysis or inspections.
- In the areas excavated after November 2014 and in the areas where any significant changes are identified, the inspector acquires new 3D data using stop-and-go scanning, which will be integrated into the 3D model acquired in 2014 in order to obtain an updated as-built 3D model.
- New or updated drawings received from the operator will be verified against the updated as-built 3D model.
- The updated 3D model will be stored on site and serve as a reference for subsequent inspections.

The effort for the 2015 DIV will be considerably smaller than in 2014 while maintaining an accurate and up-to-date 3D model of the complete repository. If the procedure is repeated in subsequent inspections, it will enable the inspector to efficiently and effectively verify the correctness of the provided design information and to assure that no undeclared modifications to the facility occurred. It will also allow to navigate and authenticate the position in a facility which will become larger and more complex as the excavations advance.

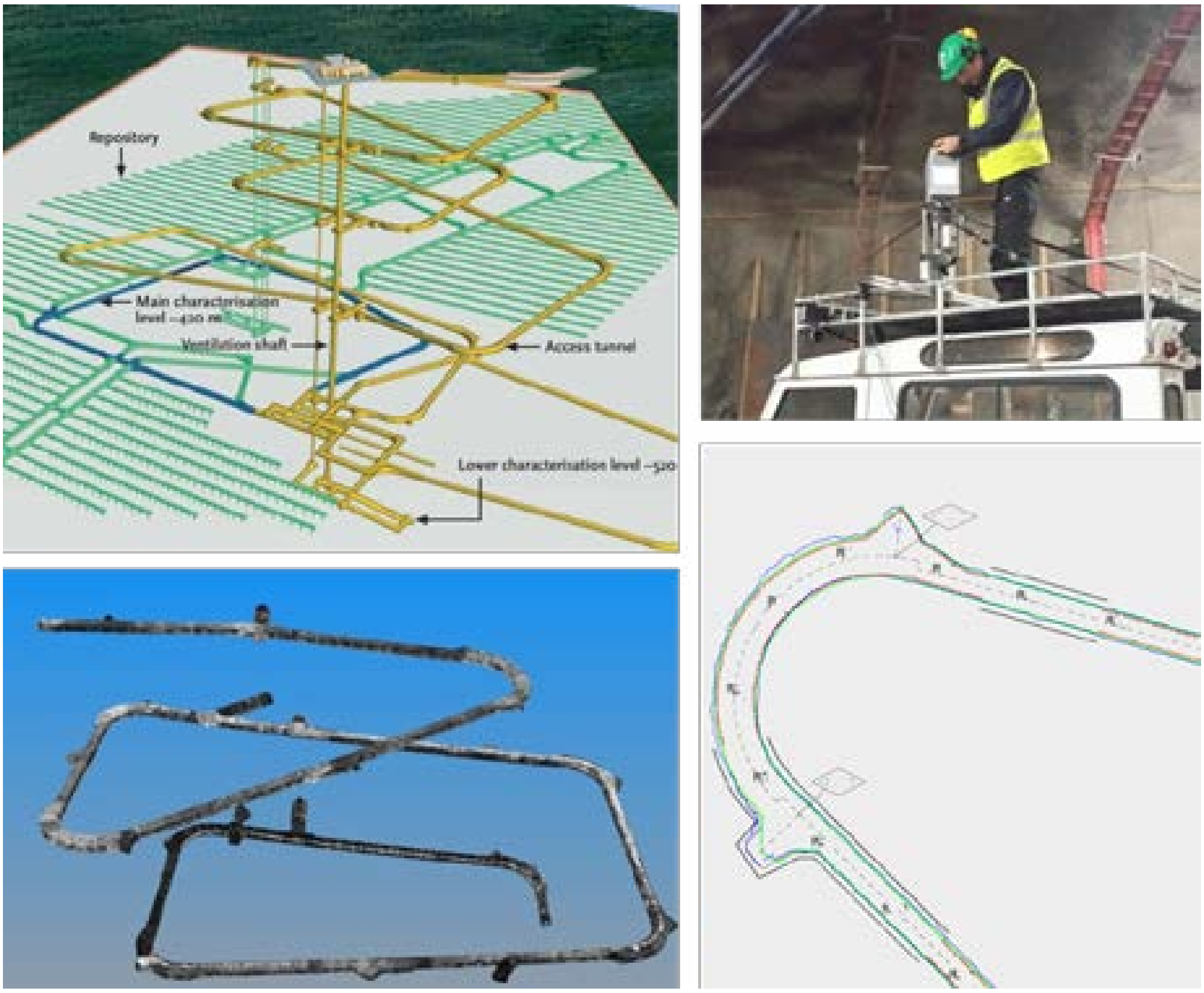


Figure 9: The top left image shows a schematic drawing of the ONKALO repository (yellow corresponds to excavations existing in 2014; green corresponds to deposition tunnels to be excavated in the future. Image courtesy of POSIVA). Data acquisition for the 2014 DIV at ONKALO was carried out in stop-and-go mode. Two scanners were mounted on a car roof (top right image) and two were mounted on a tripod. Over 900 scans were acquired in order to generate an as-built 3D model of the complete repository (the bottom left image shows a model of the first two km which was acquired during a technology demonstration in 2007). A cross section of the 3D model was generated and used to verify the drawings provided by the operator. The bottom right image is the zoom of a sample drawing; the blue line is a cross-section of the as-built laser data, which was used to verify the CAD drawing.

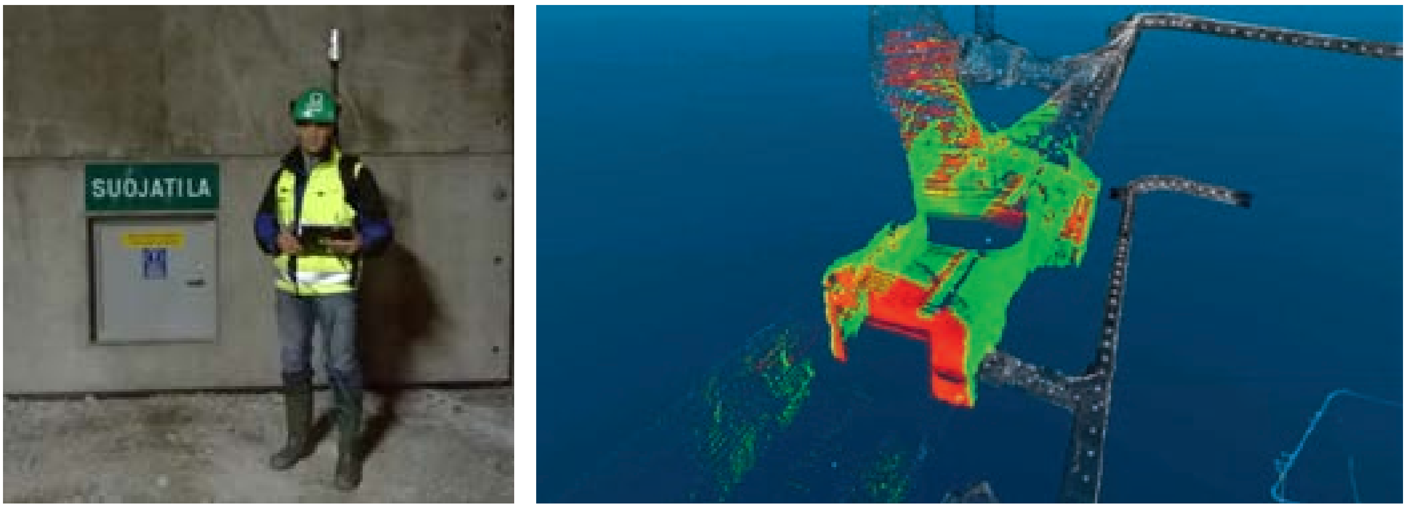


Figure 10: *Left:* MLSP demonstration in ONKALO in 2014. *Right:* Snapshot of the MLSP interface as it is provided to the user in real-time. The model acquired during the 2007 demonstration (which is used as reference) is shown in grey. The data acquired during the 2014 demonstration is color-coded as follows: green corresponds to objects that already existed in 2007 (i.e. the main tunnel excavation); red corresponds to changes (e.g. the fire door that was constructed after 2007).

4.2 Indoor Localisation

Accurate indoor localisation of the inspector increases the efficiency and effectiveness of safeguards activities. It allows i) verifying the position and layout of a facility and ii) associating all observations and measurements made during the inspections with the respective location. Location-tagging the data acquired during the inspection greatly facilitates subsequent reporting, analysis and future inspections. For this reason, the IAEA tool kit of portable instruments, which is used to support complementary access activities, includes a GPS instrument. However, GPS measurements are not available indoors and therefore safeguards inspectors currently have no means to accurately localise themselves inside nuclear facilities.

As shown above, mobile laser scanning can be used for localization in indoor environments. Depending on the availability of a 3D reference model, it can be operated in two different modes:

1. *3D reference model available.* If a 3D reference model of the indoor environment is available, mobile laser scanning provides real-time localization with centimeter accuracy as described in section 3.The reference model can be acquired with stop-and-go scanning. However, this might be a considerable effort, depending on the size and complexity of the environment. In some cases (e.g. at ONKALO as described above), a 3D reference model has already been acquired for DIV purposes.
2. *3D reference model* ***not*** *available.* In cases where it is not practicable to acquire an accurate reference model prior to the inspection, the mobile laser scanner can be operated in an *Odometer* mode, i.e. the location is tracked (with reduced accuracy and robustness) using only the data acquired with the mobile scanner. After the inspection, the mobile 3D data can be re-processed using a SLAM (Simultaneous Localisation and Mapping) approach, which globally optimizes the generated track and generates a 3D model from the acquired data. Hence, inspector observations can be accurately located retrospectively and the 3D model can be used as reference for subsequent inspections. SLAM processing for the MLSP is currently under development. [5]

In 2014, the IAEA organised a technology workshop which aimed to evaluate the performance and suitability of currently available indoor navigation systems [6]. It defined the following use-cases:

- *Position Authentication*: “Inspector quickly verifies that he has been taken to the expected site location”.
- *Mapping*: “Inspector decides to perform an overall site survey; he walks or drives through the site so he may confirm / complete the IAEA knowledge of the site.”
- *Tracking and navigation*: “While surveying, the inspector continuously traces his itinerary through the site; he may also navigate toward a specified location.”
- *Geo-tagging/Location-based services*: “During the site survey, the inspector writes notes, draws sketches, records audio notes, takes pictures, takes samples, and makes various measurements. The inspector can also review on the map the actions conducted during the previous inspections.”

For the technology evaluation, IAEA defined a set of scenarios, which aimed to simulate different situations where the inspectors need to position themselves and navigate inside vast and complex sites. The evaluation was carried out inside and outside the IAEA HQ in Vienna. In total, eleven systems were tested, which can be grouped in two

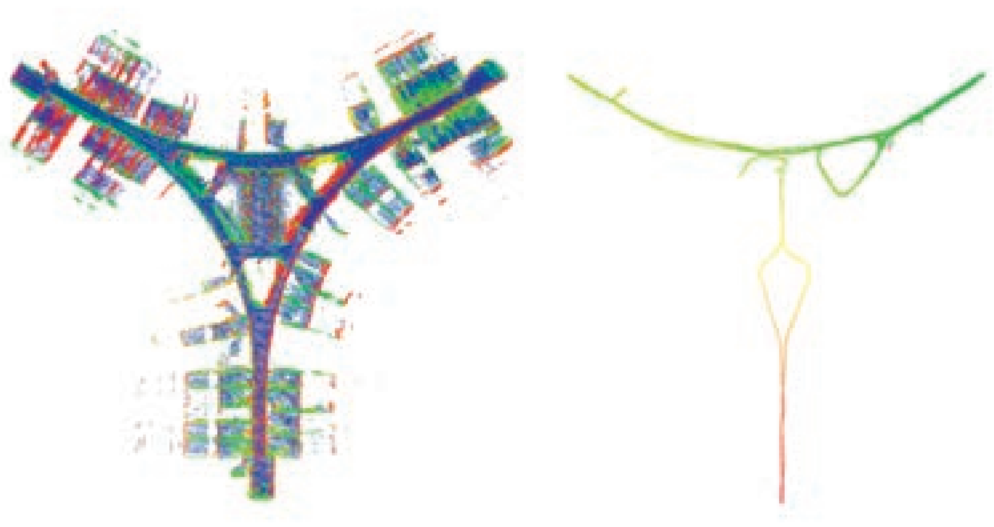


Figure 11: Results obtained with the MLSP for one of the scenarios of the IAEA technology evaluation workshop, in which the user walked through a corridor of the IAEA HQ. *Left:* top view of the 3D model created from the acquired 3D data. It covers the corridor and some of the offices (where the doors were open). *Right*: the track followed by the user as computed by MLSP.

categories: i) systems using laser scanning, such as JRC's MLSP and ii) systems based mainly on MEMS (Micro-electro-mechanical-systems) sensors including inertial measurement systems (IMU), magnetometers and compass. The JRC participated in the evaluation using an early version of the MLSP. Figure 11 shows the result of one of the scenarios.

In its final report, the IAEA concludes that "*laser-based sensors are the only solutions today that can offer near-perfect constant accuracy*". Due to operational issues (i.e. the size of the system), IAEA does currently not envisage laser-based systems to be part of the standard inspector equipment, "*however they could become extremely valuable tools for specific missions*" [7].

4.2.1 Localisation Accuracy

In April 2015, JRC participated in an indoor localization competition in order to confront the MLSP accuracy to other state-of-the-art systems. The competition, organised annually by Microsoft, gathers teams from industry and academia to evaluate the performance of their respective localisation systems. The competition is carried out in two categories, involving infrastructure-based systems which require installation of equipment such as radio emitters in the environment and infrastructure-free systems which rely only on sensor readings. The JRC competed in the infrastructure-free category, where it came first, with a localisation error of 0.2 m, which also surpassed the best result in the infrastructure-based category, in which the winner had a localisation error of 0.31 m (see [8]).

The competition evaluated the measured 2D position of the user at pre-defined markers, i.e. the results were influenced by the accuracy with which the user positioned himself on the markers. The actual accuracy with which the MLSP sensor can be located in 3D space corresponds to the accuracy of the laser scanner, i.e. approximately 0.02 m.

5. Future Activities

IAEA and ENER plan to use the MLSP for future DIV/BTC verifications at the ONKALO underground repository in Finland as described above and it might be similarly be applied at other facilities.

JRC will further develop the system to make it more applicable for indoor localization during complementary access activities. Inter alia, JRC plans to i) implement a SLAM approach so that MLSP can be used without prior availability of a 3D reference model and ii) provide an interface to easily integrate with other in-field tools, location-based applications and HQ infrastructure.

Furthermore, MLSP will benefit from related technology advances: i) the miniaturization of 3D sensors will continue and therefore the overall size and weight of the system will reduce and ii) further developments in augmented and virtual reality technologies (such as Google Glass) will complement MLSP's accurate 3D positioning capability to provide enhanced infield inspection tools.

6. Conclusion

The paper describes the Mobile Laser Scanning Platform (MLSP), which was developed at the JRC for nuclear safeguards applications. MLSP is a portable system that combines a mobile 3D laser scanner with on-board processing for real-time localization, tracking and change analysis.

Mobile laser scanning can complement traditional stop-and-go laser scanning for DIV/BTC verification, thus significantly reducing the required acquisition and processing time. IAEA and ENER intend to use mobile laser scanning for future DIV/BTC verification at the Finnish underground repository to verify that no undeclared modifications to the facility have occurred.

Indoor localization is an enabling technology for many location-based applications and facilitates the storage, analysis and retrieval of observations and measurements made by an inspector, for example in complementary access scenarios. Mobile laser scanning provides an indoor localization accuracy which is currently not achievable with any other technology and therefore has the potential to significantly increase the inspector's efficiency and effectiveness during specific missions such as complementary access inspections. Future developments will further increase the applicability of 3D-based indoor localization for nuclear safeguards.

References

[1] E. Agboraw, S. Johnson, C. Creusot, S. Poirier, H. Saukkonen, B. Chesnay and V. Sequeira, "IAEA experience using the 3-Dimensional Laser Range Finder," in *IAEA Safeguards Symposium: Addressing Verification Challenges*, Vienna, 2006.

[2] P. Chare, Y. Lahogue, P. Schwalbach, A. Smejkal and B. Patel, "Safeguards By Design – As applied to the Sellafield Product and Residue Store (SPRS)," *ESARDA Bulletin (46),* pp. 72-78, 2011.

[3] C. Sanchez, P. Taddei, S. Ceriani, E. Wolfart and V. Sequeira, "Localization and Tracking in Known Large Environments using Portable Real-time 3D Sensors," *Computer Vision and Image Understanding,* Submitted.

[4] Y. Chen and G. Medoni, "Object modelling by registration of multiple range images," *Image Vision Computing 10 (3),* 1992.

[5] P. Taddei, C. Sanchez, S. Ceriani, E. Wolfart and V. Sequeira, “Pose Interpolation SLAM for Moving 3D Lidar Sensors,” in *International Conference on Intelligent Robots and Systems*, Hamburg, 2015 (Submitted).

[6] IAEA, “User Requirements - Technology Evaluation Workshop of Core Components of an Autonomous Navigation and Positioning System for Safeguards (SG-UR-12711),” IAEA, Vienna, 2014.

[7] IAEA, “Report - Technology Evaluation Workshop - Autonomous Navigation and Positioning Systems - Results (SG-RP-13057),” IAEA, Vienna, 2014.

[8] Microsoft Research, “Microsoft Indoor Localization Competition - IPSN 2015,” [Online]. Available: http://research.microsoft.com/en-us/events/indoorloccompetition2015/. [Accessed 30 04 2015].

Monitoring Uranium Mining and Milling using Commercial Observation Satellites

Lalitha Sundaresan[1], Chandrashekar Srinivasan[1] and Bhupendra Jasani[2]

1. Visiting Professors, International Strategic and Security Studies Programme, National Institute for Advanced Studies (NIAS), Bangalore, India
2. Department of War Studies, King's College London, London, UK

E-mail: sundaresan.lalitha@gmail.com, chandrashekar.schandra@gmail.com, bhupendra.jasani@kcl.ac.uk

Abstract:

All the states that have signed the Additional Protocol to their Safeguards Agreements with the International Atomic Energy Agency (IAEA) will need to submit description and information specifying the location of their nuclear fuel cycle activities, including their operational and shut down uranium mines. While satellite imagery is useful for monitoring changes in the declared nuclear facilities, there has not been much discussion of using this imagery to monitor the early part of the nuclear fuel cycle namely uranium mining and milling.

The availability of satellite data cost free on the Google Earth web site and commercially from various imagery providers makes it possible for analysts to make assessments concerning the nuclear fuel cycle activities of various countries of interest. The mining of uranium and its conversion through a milling process into U_3O_8 (yellowcake) is the first step of a complex conversion cycle that determines how the mined material will be used.

Our study discusses the use of satellite imagery for identifying and monitoring uranium mining and milling activities. In the study an attempt is made to answer the following questions:

1. Can we identify uranium mines using openly available satellite imagery?

2. Can we use various steps in uranium milling operations to identify such mills across the world?

3. Are there other extraction processes that share similar features with those for uranium? If so, then are there any special features present or absent in the sequence of operations for their extraction that helps an analyst separate a uranium operation from other operations that share some or all of the features present in the extraction of uranium?

Based on empirically derived observables and signatures from satellite imagery for typical uranium extraction operations we have derived a decision making algorithm for determining whether a particular facility can be categorized as a uranium mill or whether it should be categorized as some other facility.

The method has been used to look at some copper mills across several locations and have shown that the decision making algorithm does help us to separate out a uranium mill from a copper mill.

Keywords: Uranium mills, Fuel cycle, Spatial features, Uranium mines, International Safeguards, Satellite Images.

1. Introduction

The need to prevent nuclear weapons proliferation has been of serious concern for the last several decades. These concerns resulted in a number of bi-lateral and international arms control treaties. The treaty on the Non-Proliferation on Nuclear Weapons (NPT) was one such international agreement under which the parties undertook to limit the spread of nuclear weapons and related technologies by a series of measures while encouraging the peaceful uses of nuclear energy under international safeguards system implemented by the International Atomic Energy Agency (IAEA).

The safeguards system is used to verify compliance with the NPT through inspections conducted by the IAEA. While this system worked well for the declared nuclear activities of a party to the NPT, it became apparent that it was difficult for the Agency to detect undeclared nuclear activities. Thus, the Director General's Standing Advisory Group on Safeguards Implementation (SAGSI) recommended that, as one measure "assessment of the usefulness, technical feasibility, associated costs and acceptability of the Agency obtaining satellite photographs from commercial sources" should be carried out (SAR-17, Report to the Director General on the 38th Series of SAGSI meeting, 21-22 March1994). Eventually in 1997, the new safeguards Model Protocol Additional to the Agreement(s) between State(s) and the International Atomic Energy Agency for the Application of Safeguards, INFCIRC/540, was established. The implementation of this provides the Agency with the capability to detect undeclared materials and activities in a state. The use of open sources, commercial satellite imagery, further additional information and extended access to nuclear facilities and other locations gave the Agency credible assurance on the absence of undeclared nuclear materials and activities.

The IAEA gathers and analyses safeguards relevant information about a State from: (a) information provided by the State party to the safeguards agreement, (b) safeguards activities conducted by the Agency on the ground and (c) from open sources and third parties [1]. The IAEA's analyses consist of validation of information provided by the States against information collected by the Agency under (b) and (c) including that obtained from commercial satellite imagery. Information may differ depending on whether it is acquired under a comprehensive safeguards agreement (CSA), under a CSA together with the Additional Protocol Agreement (APA) or that obtained on a voluntary basis.

Under the Additional Protocol, the Signatory States have to provide "Information specifying the location, operational status and the estimated annual production capacity of uranium mines…" [2]. This has increased the amount and type of information that states will have to provide to the IAEA. At the same time, the verification workload of the IAEA inspectorate has also increased commensurately. Keeping in mind the security, or the lack of it, in the world in recent years, the IAEA is bound to find itself in a situation where physical verification of the declared nuclear facilities will become increasingly difficult, which will also include the expanded list of declarable facilities, including those that form the early part of the nuclear fuel cycle, e.g., uranium mining and milling facilities. Newer remote monitoring methods and technologies, such as non-intrusive commercial satellite-based imaging, can strengthen nuclear safeguards by making the IAEA's verification process more efficient [3].

In 2001, the IAEA's Satellite Imagery Analysis Laboratory (SIAL) became fully operational [4] and the Agency has been using commercial satellite images as a tool for safeguard purposes routinely and it has become one of the most important information sources that the IAEA's Department of Safeguards has for remotely monitoring nuclear sites and activities [5, 6]. With the Additional Protocol, that monitoring now also applies to the early part of the nuclear fuel cycle which includes uranium mining and milling. However, satellite imagery has not been used in a major way by the IAEA for looking at existing or newly created mining or milling operations and assessing whether they are used for the production of uranium.

The results of what can be learnt about uranium mining and milling using satellite imagery have been published in the open literature. For example spatial features associated with uranium extraction process are described in the Photo Interpretation Handbook [7]. During the Cold War period, the CIA monitored the uranium mining and milling activities in the USSR and Eastern European countries. Even as early as 1959, the CIA attempted to estimate production of uranium oxide in the Pyatigorsk Mill, USSR based on the ore grade and the size of the tailings pond as seen on aerial photography [8]. With the launch of the CORONA satellite in 1960, low resolution satellite images became available for intelligence gathering. Estimates of uranium production of the Steiu Plant in Rumania was however, not possible with this image based on the size of the tailings pond, because of the low resolution [9].

A few studies have been carried out to assess the effectiveness of high resolution satellite images as a tool for verification of safeguards agreements between the IAEA and various countries [10, 11, 12, 13, 14, 15, 16]. Some of these efforts try to define key features of a nuclear facility and seek to uniquely identify them in a satellite image. Evaluation of specific spectral and temporal characteristics of newer satellite sensors is also being carried out. Use of high resolution SAR and optical data for 3D analysis in the context of Safeguards has been of interest more recently [12]. Automated object based image analysis methods (involving the spectral signature, size, shape, proximity aspects of facility components) have been found to be particularly useful as demonstrated by Nussbaum and Menz [17].

The present paper is an effort to demonstrate the applicable aspects of commonly available satellite imagery for identifying and monitoring uranium mining and milling activities. Towards this it seeks to answer the following questions:

- Can we use the various steps in uranium milling operations to identify such mills across the world using commercially available satellite imagery?
- Are there other extraction processes that share similar features with uranium extraction processes? If so, how do we distinguish uranium mills from these mills in a satellite image?
- Is it possible to make an assessment of the uranium production capacity of a mill identified in a satellite image?

2. Past Work

A number of studies have reported the difficulties in uniquely identifying uranium mining and milling activities since the concentration of uranium is rather low at most places and does not show spectral characteristics that will help to uniquely identify it in a satellite image [18, 19, 20, 21]. The steps involved in the conversion of uranium ore to yellow cake were used to develop a set of keys to identify a uranium mill in a high resolution hyper-spectral satellite image. These studies demonstrated that the potential observables which are present in the uranium mining and milling operation, but not in copper mining and milling, include the discriminator station, pyrolusite (manganese dioxide) which is used as an oxidant in leaching, the pregnant uranium leach liquor produced in the sulfuric acid leaching process, the concentrated uranium strip solution generated from solvent extraction, and finally the yellowcake produced from the precipitation and drying steps. Most of these features do not have unique spectral signatures and their identification is further complicated by their small spatial extents. The Discriminator station or the Radiometric sorters perhaps could be identified with high spatial resolution data available these days.

Using the Ranger mine and mill as an example, researchers at the Sandia National Laboratory analysed the potential use of multi-spectral as well as hyper spectral data from a number of remote sensing satellites to separate out any unique features of a typical Uranium mining and milling operation [22]. Apart from magnesium chlorite the only other identifiable signature came from the Sulphur heaps at the Ranger site which is used to manufacture sulphuric acid for the acid leaching process at Ranger. The study concluded that hyperspectral data could not distinguish between uranium processes from other milling processes such as that of copper, zinc, vanadium, phosphorous and Rare Earths. Further the study pointed out that while high spatial resolution satellite systems such as Quickbird lack sufficient spectral resolution to uniquely identify many materials, spatial information provided by these systems could complement information obtained from high spectral resolution systems such as Hyperion. A unique aspect of this study however, was the creation of a decision tree that linked each step in the milling operation at Ranger to similar processes used in the extraction of other materials of commercial and strategic importance.

An important conclusion that emerges from these studies is that it is difficult to identify a uranium mill using only spectral signatures be it multi spectral or hyper-spectral satellite images. Perhaps in the future, with higher spatial resolution hyper-spectral imagery, such discrimination of sulphur heaps and uranium ore piles might be possible. It is also recognized that a combination of the hyper spectral images along with radar images are definitely useful to monitor the activities of a milling site and record changes that may be happening for various purposes, clandestine or otherwise [23].

3. Our Approach

As the commercial satellite images are expensive, we have largely relied on the images published cost free on the Google Earth (GE) web site.

While it is recognized that the IAEA would require the latest data, it could use GE images to study the historical development of a particular site.

In our approach a set of keys for identification of a uranium mill is developed based on the spatial features of the equipment used in the milling operations instead of looking for spectral signatures.

This is achieved by interpreting the GE images of a large number of commercial uranium mills across the world.

A comprehensive understanding of the spatial signatures of the uranium operations at each site is built up using the process flow sheets of the mill along with publicly available information about the mill.

Together with the Google Earth (GE) image of the mill, the keys for identification are developed.

The most commonly occurring features in the sample sets along with their signatures are then used to decide whether a mill seen on a satellite image is a uranium mill or not.

4. Uranium Milling Process

The process of uranium extraction is very well known [24]. However, to integrate it with our study, a schematic of a typical process for the extraction of uranium from its ore is shown in Figure 1.

The associated equipment / reagents with each of these steps are also shown in the figure.

Our objective is to determine which of the equipment are unique to a uranium milling operation and visible and identifiable in a satellite image.

For the purpose of this study we have not considered those mills that use heap leaching as the only method for leaching.

The reason for this omission is that the process steps involved in this case will differ slightly and it may not be possible to uniquely identify such mills in a satellite image.

We selected 11 uranium milling operations and our sample set is shown in Table 1.

The imagery available on GE for each of these mills was studied in detail along with other publicly available information.

The set of observables that we could identify from these images formed the basis for identifying the key observables needed to identify a uranium mill.

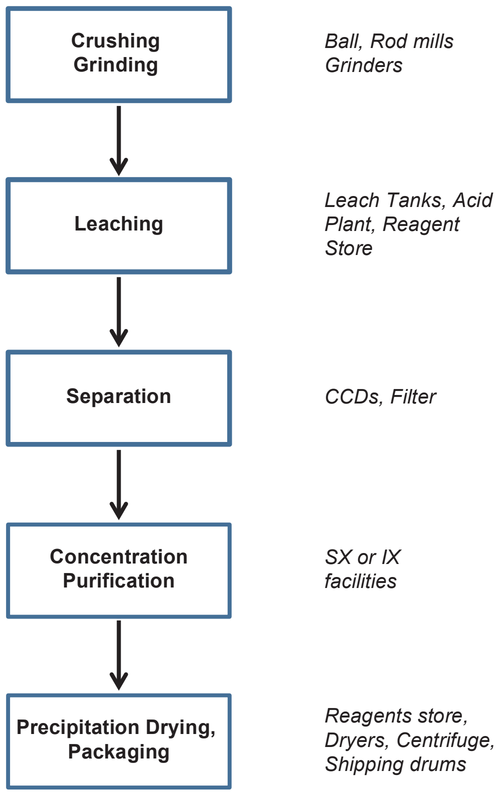


Figure 1: Simplified Overview of the Steps involved in Uranium Milling Process

Country	Mill Name	Location (Lat / Long)	Owner	Start Year
USA	Sweet Water	42 03 N 107 54 W	Shut Down	1981
Canada	Rabbit Lake	58 15 N 103 40 W	CAMECO	1975
Australia	Ranger	12 41 S 132 55 E	ERA	1981
Canada	Mclean Lake	58 21 N 103 50 W	Areva	1999
Canada	Key Lake	57 13 N 105 40 W	CAMECO	1983
Niger	Arlit	18 47 N 7 21 E	Areva	1970
Namibia	Rossing	22 28 S 15 03 E	Rio Tinto	1976
Namibia	Langer	22 49 S 15 20 E	Paladin	2006
Russia	Krasnokamensk	50 06 N 118 11 E	Argun	1968
Czech Republic	Rozna	49 30 N 16 14 E	DIAMO	1958
Romania	Feldiora	45 50 N 25 30E	State Owned	1978

Table 1: Sample set of Uranium Mills

5. Observations from Satellite Images of a Uranium Mill

The uranium mill features observable in a satellite image for the sample sites are summarised in Table 2

Though crushing, grinding and slurry preparation facilities are identifiable in most of the images they do not offer any special features associated with only a Uranium Milling operation.

Radiometric sorters are used in many Uranium mills to improve the ore quality. While they can be identified in the satellite images of some of the mill or mine sites, we could not identify them in all the mills or mines of our sample set.

The most commonly visible feature in the satellite image is the Counter Current Decantation (CCD) unit, used in the solid / liquid separation process. Figure 2 shows some typical CCDs of some of the mills. In all the cases except the Sweet Water mill, this feature is easily identifiable. The Sweet Water mill was closed down in 1984 and according to published reports the CCD is housed inside a building.

There are several features associated with the leaching process. Some feature or the other is seen in all the mills.

	Acid Plant	Sulphur store	Acid/Alkali store	Hot Leach	Leach tanks	CCD	SX	IXColumn	NH_3 tanks
Sweet Water	NA	NA	S	NS	NS	NS	Building?	NA	S
Rabbit Lake	S	S	S	NS	S?	S	Building?	NA	S
Ranger	S	S	S	NS	S	S	Pattern seen	NA	S
Mclean Lake	S	S	NS	NS	NS	S	Building?	NA	S
Key Lake	S	S	S	Smoke	NS	S	Building?	NA	S
Arlit	S	S	S	NS	S	S	Pattern Seen	NA	S
Rossing	S	S	S	NS	S	S	Pattern Seen	S	NS
Langer Heinrich	NA	NA	S	Heat Exch.	S	S	NA	S	NA
Krasnokamensk	S	NS	S	Chimney Seen	Autoclave	S	NA	S	NS
Rozna	NA	NA	S	Smoke	NS	S	NA	NS	NS
Feldiora	NA	NA	S	Chimney seen	Autoclave	S	NA	S	NS

S – Seen, **NS** – Not Seen, **NA** – Not Applicable

Table 2: Uranium Mill Features Observable in a Satellite Image

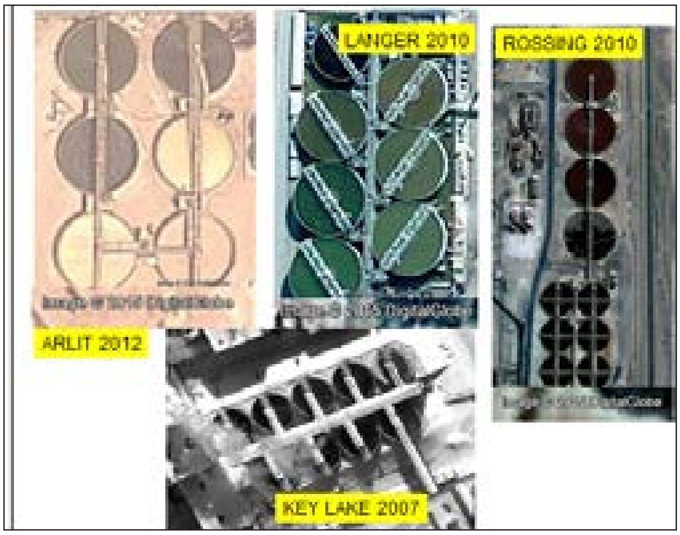


Figure 2: CCD units as seen in a Google Earth (GE) satellite image (Key Lake Image was obtained separately from DigitalGlobe)

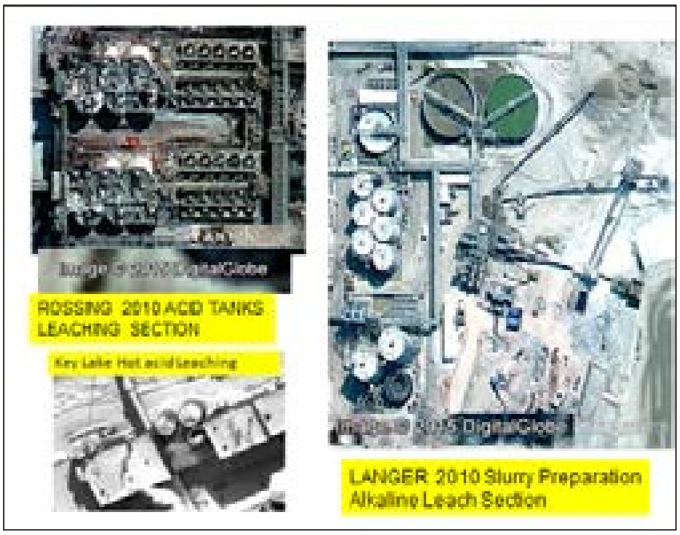


Figure 3: Leaching equipment as seen in Google Earth satellite image (Key Lake Image was obtained separately from DigitalGlobe)

Of the 11 mills Langer Heinrich, Rozna and Feldiora use alkaline leaching, while the other mills use acid leaching. Since alkaline leaching involves higher temperatures; one can look for evidence of chimney, heat exchangers or even smoke. Additionally in the case of acid leaching one can see either the acid plants or the leach tanks and sometimes the acid storage tanks close to the leaching facility.

Figure 3 shows typical leach tanks and leaching sections of some of the mills in our sample.

Unlike the CCDs, the leaching facility is difficult to identify and requires knowledge of the process being employed in the mill. However, we do know that the leaching operation follows the ore preparation step and is followed by separation and therefore the sequence of operation helps to identify some of the leaching features.

The next feature of interest is the equipment associated with the process of concentration and purification. In most mills this is done using either the solvent extraction (SX) columns or the ion exchange (IX) process. Occasionally a combination of both may be used.

The SX columns are housed inside a building and thus not readily identifiable. In our sample mill sites we, however noted that the SX columns are housed inside a sequence of identical buildings and linked to these are the storage tanks containing the solvents used in the SX process (Figure 4).

The IX columns are usually left in the open and are visible in the satellite image (Figure 5).

The features associated with precipitation, drying and calcining are not uniquely identifiable in a satellite image. In most cases they have to be identified indirectly by the presence of containers holding solvents and reagents used for this purpose. Proximity to the SX or IX facilities of

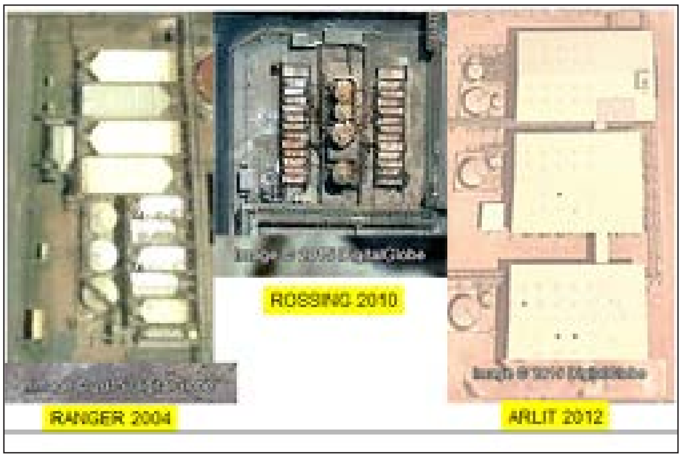


Figure 4: Solvent Extraction Buildings as seen in a Google Earth satellite image

such features is another aspect that we can use to identify this facility. In some of the mills where ammonia is used, the ammonia cylinders are seen clearly in the satellite image.

To summarise the procedure for identifying a uranium mill from a GE image, we first identify the CCD circuit; then try to locate the leaching facility upstream. If the CCD process is followed by a SX or IX facility, we could conclude with high level of confidence that the facility is a uranium mill.

This approach has certain limitations because many other mineral extraction processes are very similar to the uranium extraction process. For instance the process steps of copper, zinc and vanadium are very similar to Uranium. Of these it is most difficult to discriminate copper and uranium extraction processes spectrally in a satellite image.

By identifying spatial features that are unique to copper mills, we will be able to differentiate a uranium mill from a copper mill.

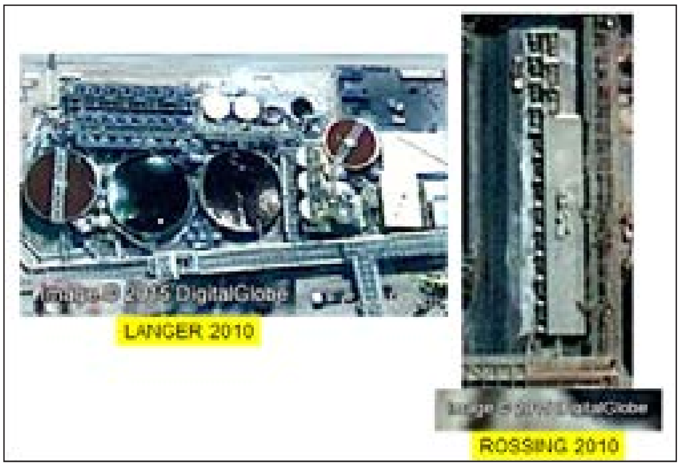


Figure 5: Ion Exchange columns as seen in a Google Earth satellite image

6. Copper Extraction Process and Observables in a Satellite Image

The major steps involved in a copper extraction process are shown schematically in Figure 6.

A major difference between copper and uranium is the scale of operation. Invariably due to economic considerations, the copper processing facility will be several times larger than the uranium operation.

Copper occurs mostly in the Sulphide or Oxide forms. While the crushing and grinding steps are common to all extraction processes, the process steps in the case of sulphide ore are different from that of the oxide ore. This is shown in the Figure 6.

The sulphide ore goes through a froth flotation process after the initial crushing and grinding which concentrates the copper part. The froth from the flotation process contains the bulk of the copper. The froth is dried and then sent directly to a smelter. The smelter may be located at the mill site or may be located elsewhere. The smelter converts the copper concentrate into blister copper which is further refined to produce anodic copper and finally goes through an electro winning step to produce high purity copper.

The tailings from the froth flotation may also contain copper which could be recovered. These tailings are leached with sulphuric acid, passed through a series of CCDs followed by a solvent extraction step. The copper solution that comes out of the solvent extraction step is then sent to an Electrowinning Facility for the extraction of copper.

Thus a mill which processes low grade copper ore or a part of a copper mill which processes the tailings from a froth flotation process will look similar to a uranium mill. It will have the features such as CCD circuits, SX units in addition to the acid leach facilities that we have seen in a uranium mill.

However, the differentiating factor for the extraction of copper from flotation tailings is that after solvent extraction it goes to an electro winning facility instead of a precipitation facility. Since such an electro winning facility has a typical signature evidence of this step in a satellite image can be used to separate out a Uranium mill from a copper mill.

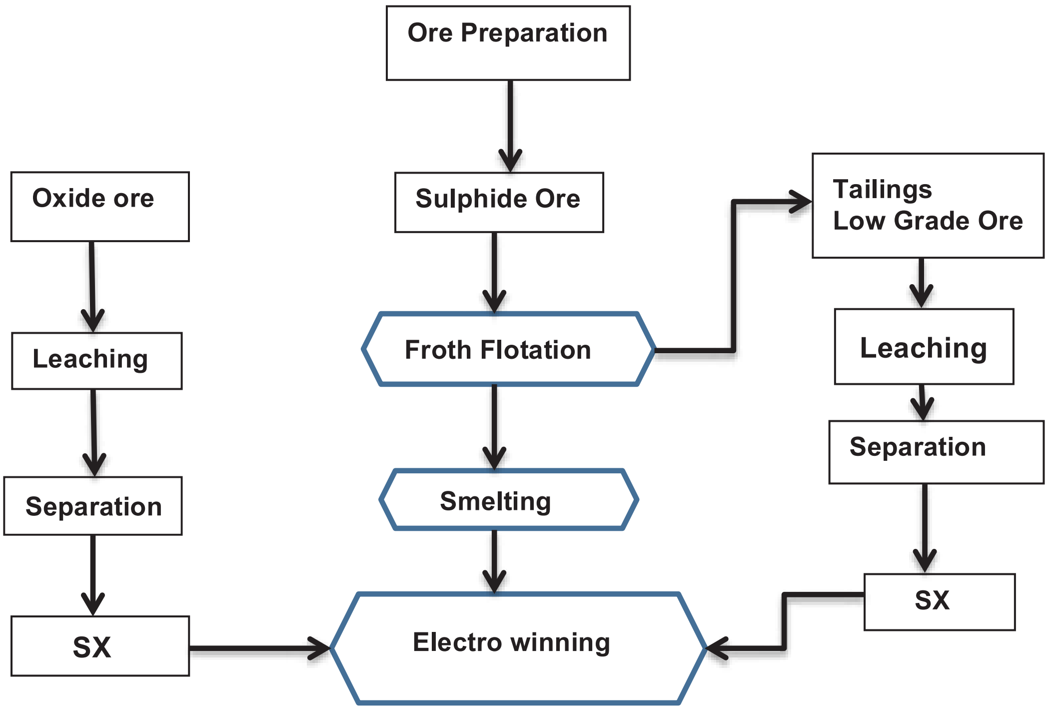


Figure 6: A Simplified Diagram showing the Copper Extraction Process Steps

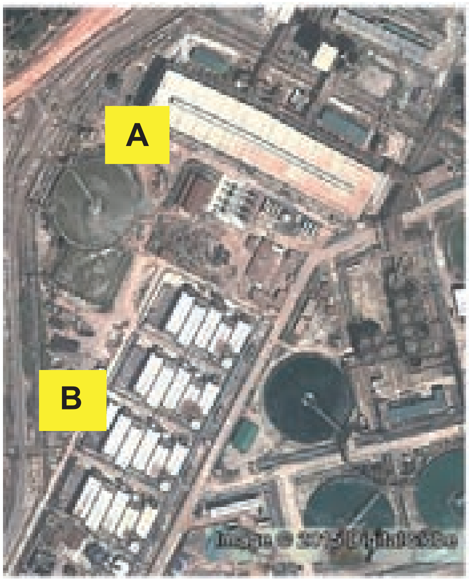


Figure 7: Google Earth image of Nchanga Copper mill (A – Electrowinning, B – SX)

Figure 7 shows a typical electro-winning facility as seen in a satellite image.

In the figure the long building (A) is an electro winning facility which can be easily identified and this is co- located with the solvent extraction facility in the foreground (B).

Copper occurring in the oxide form is typically leached using sulphuric acid after suitable crushing and grinding. Following concentration through a solvent extraction process the solution containing copper is sent to an electro-winning facility. Depending on the concentration of the ore the leaching step may also be followed by a CCD sequence prior to solvent extraction and electro-winning.

Again the differentiating step between copper and uranium is the electrowinning facility.

7. Key Differentiators for a Uranium Mill

The sequence of Acid or Alkaline leaching – CCD – solvent extraction – precipitation is typical of all Uranium mills.

The CCD unit of these mills is the most amenable to observation from satellite. Though its absence does not completely rule out Uranium, its presence is a robust indicator of a potential Uranium milling operation.

The leaching step is the next most visible feature in a satellite image. Both direct and indirect signatures are available to make inferences about this step. The absence of a leaching process rules out a Uranium mill.

Thus the sequence of CCD preceded by a leaching step provides a baseline signature for a possible Uranium Mill.

In many cases solvent extraction facilities have features such as repetitive identical buildings close to the CCDs that can be identified through satellite imagery.

Ion exchange facilities can be seen in a satellite image unless in rare cases they are housed inside buildings.

In the case of precipitation, storage tanks for the various chemicals and their location in the flow of material provide some indications. Ammonia tanks used in many cases for the precipitation of Uranium are often identifiable in a satellite image. Along with a CCD and a leaching step Ammonia tanks provide a firm indication of a Uranium extraction operation.

Since the solvent extraction or ion exchange or even the precipitation steps in a Uranium mill do not provide very robust signatures one way to enhance the reliability of our classification is to eliminate other materials that share the Leaching - CCD - Solvent Extraction sequence.

Copper extraction mills that may in some cases share a similar Leaching – CCD – Solvent extraction sequence can be eliminated by the presence of Electro-winning, Smelting and froth flotation facilities in such extraction processes. All of these have clear signatures and can be identified easily in a satellite image. Through such elimination of various alternatives that share the leaching step and in some cases the CCDs as well as solvent extraction steps, we can increase the probability that the mill we are seeing is indeed a Uranium Mill.

8. Assessment of Production Capacity

Using the observables from the satellite image such as the number of CCD circuits, the diameter of the CCD in a mill along with the average ore grade, we have been able to arrive at an empirical equation to estimate the production capacity of the mill.

The equation was derived linking the nominal production capacity data of the sample mills in our study with the measurements made on the satellite images of these mills.

The equation is in exponential form:

$$P = k * G^a * N^b * A^c$$

Expressed in log form and estimating the coefficients k, a, b and c using the sample data gives,

$$Ln\ P = 3.112976 + 0.457613*LnG + 0.956309*Ln\ N + 0.561587*Ln\ A$$

Where,

k = Constant
G = Ore grade in percentage U_3O_8
N = Number of CCDs
A = Area of the CCD in meter square.

The data used for this purpose is shown in Table 3.

The nominal capacity for the mills is taken from the Red Book.

The estimated production values for the mills from the empirical equation are also shown in the table for comparison.

The results are reasonably good except for the Russian Mill.

Agencies such as IAEA having access to more accurate data will be able to improve upon these estimates.

Country	Mill Name	Ore Grade (% U_3O_8) G	CCD Nos. N	CCD Diameter (meters) D	Nominal Production Capacity (tonnes) P	Predicted Capacity (Tonnes)
USA	Sweet Water	0.048	6	9.752782825	350	401.41
Canada	Rabbit Lake	0.79	4	30.00530739	4615	3467.43
Australia	Ranger	0.13	8	34.65020841	4660	3463.12
Canada	Mclean Lake	1.22	8	12.85019021	3077	3166.65
Canada	Key Lake	3.40	8	20.00353826	7200	8320.77
Niger	Arlit	0.30	6	23.00650697	2330	2434.56
Namibia	Rossing	0.03	10	56.32028518	4000	3781.54
Namibia	Langer	0.05	7	23.15041609	1425	1251.39
Russia	Krasnokamensk	0.18	6	52.01257401	3000	4817.18
Czech Republic	Rozna	0.378	5	24.98407136	3200	2493.75
Romania	Feldiora	0.12	4	28.00705098	1120	1354.75

Table 3: Data from the Sampled Mills (All data taken from Uranium 2009: Resources, Production and Demand, A joint Report by OECD NE Agency and IAEA, 2010, commonly called the Red Book)

This estimation process is applied to an Indian mill at Turamdih, Jharkhand.

This mill uses acid leaching and ion exchange. (See Figure 8).

The mill processes uranium ore of grade 0.034%. In the satellite image we can identify 3 CCDs of diameter 13m.

Using the empirical equation, we estimate the production capacity of the mill to be 244 tonnes which compares well with the nominal capacity of 190 tonnes.

Figure 8: Google Earth Image of Turamdih Mill

9. Conclusion

This paper demonstrates how publicly available images from Google Earth can be a very useful research tool to identify a uranium mill. It is also a very useful tool to study the development of the mill site, as one can obtain past images.

It is possible to identify a uranium mill in a satellite image using the spatial features of the equipment used in the extraction process.

It is also possible to distinguish a uranium mill from a copper mill since the spatial features associated with the copper mill are different from that of the uranium mill. The presence of the electro winning facility in a copper mill enables us to differentiate it from a uranium mill.

An empirical equation is provided to generate a rough estimate of the production capacity of a uranium mill identified on a satellite image. The number of CCDs, the diameter of the CCD and the ore grade are the key factors used to make this estimate.

10. Acknowledgement

The authors benefited significantly from the comments received from one of the reviewers. The authors would like to thank R.Nagappa and N.Ramani for many useful suggestions while doing this study.

References

[1] https://www.iaea.org/safeguards/safeguards-in-practice/information-collection-and-evaluation

[2] INFCIRC/540, Article 2(v), Model protocol additional to the agreement(s) between state(s) and the international atomic energy agency for the application of safeguards, IAEA, Austria, Reprinted 1998.

[3] Cooley, J. N., 2006, International Atomic Energy Safeguards under the treaty on the Non-Proliferation of Nuclear Weapons: Challenges in implementation In: Avenhaus, R., Kyriakopoulos, N., Richard, M., and Stein G. (ed), *Verifying Treaty Compliance, Limiting Weapons of Mass Destruction and monitoring Kyoto Protocol Provisions*, Springer, Berlin, 61-76.

[4] "Use of commercial satellite imagery in strengthening IAEA safeguards", Chitumbo K, Robb S and Hilliard J in Bhupendra Jasani and Gotthard Stein (eds.), *Commercial Satellite Imagery – A tactic in nuclear weapon deterrence*, Springer and Praxis, 2002, pp. 23-48

[5] IAEA Information Series, *Tools for Inspection*, Division of Public Information, 04-46161/F S Series 3/03/E.

[6] Steinmaus, K., Norman, C., Ferguson, M., Rialhe, A. & J. Baute (2013): The Role of the Geospatial Exploitation System in Integrating All-Source Analysis. In: Proc. INMM 54th Annual Meeting, 14-18 July 2013.

[7] Photo Interpretation Student Handbook., 1996, National Imagery and Mapping Agency, Pp 434 -439 (http://sites.miis.edu/geospatialtools2013/files/2012/07/Photo-Interpretation-Student-Handbook.-Photo-Interpretation-Principles.pdf)

[8] Joint Photographic Intelligence Report, Uranium mining and milling complex, Pyatigorsk, USSR, Unclassified Report Published CIA, 1959, Pp16. (http://nsarchive.gwu.edu/NSAEBB/NSAEBB186/doc01.pdf)

[9] Ruffner,K.C.., 1995, CORONA America's First satellite Program, History Staff Centre for the Study of Intelligence, CIA, 1995, Washington D.C., Unclassified Report.Pp 159. https://www.cia.gov/library/center-for-the-study-of-intelligence/csi-publications/books-and-monographs/corona.pdf

[10] Jasani,B, I Niemayer, S Nusbaum, B Richter and G Stein*, International Safeguards and Satellite Imagery*, Springer Verlag, 2009

[11] Richard, J M., Paquette J P, Elbez J, New and Emerging Trends in Satellite Imagery, Presented at IAEA 2014 safeguards Symposium, (https://www.iaea.org/safeguards/symposium/2014/home/eproceedings/sg2014-papers/000042.pdf)

[12] d'Angelo, P., Rossi, C., Minet, C., Eineder, M., Flory, M. & I. Niemeyer (2014): High Resolution 3D Earth Observation Data Analysis for Safeguards Activities. In: Proc. IAEA Symposium on International Safeguards: Linking Strategy, Implementation and People, Vienna, October 20-24, 2014 (https://www.iaea.org/safeguards/symposium/2014/home/eproceedings/sg2014-papers/000163.pdf)

[13] Niemeyer I, Listner C, Canty M.(2014), Advances in the Processing of VHR Optical Imagery in Support of Safeguards Verification, Paper Presented at Symposium on International Safeguards: Linking Strategy, Implementation and People IAEA-CN220, Vienna, Oct 20-24, 2014 Session: New Trends in Commercial Satellite Imagery. (https://www.iaea.org/safeguards/symposium/2014/home/eproceedings/sg2014-slides/000273.pdf)

[14] Niemeyer, I., Nussbaum, S. Tueshaus, J. Listner, C. (2013), Advances in detecting changes at nuclear facilities using very high-resolution optical satellite imagery, Paper Presented at 54th INMM Annual Meeting, Palm Desert, USA, 14 Jul 2013 - 18 Jul 2013.

[15] Canty, M., Jasani, B., Lingenfelder, I., Nielsen, A.A., Niemeyer, I., Nussbaum, S., Schlittenhardt, J., Shamoni, M. & H. Skriver (2009): Treaty Monitoring. In: Jasani, B., Pesaresi, M., Schneiderbauer, S. & G. Zeug (Eds.) (2009): *Remote Sensing from Space. Supporting International Peace and Security. Springer*, Berlin: 167-188

[16] Jasani, B. (2004), Identification of Key Features of Nuclear Reactors for Interpretation of Images from Remote Sensing satellites, *Journal of Nuclear Materials Management*, Vol. XXXII, No. 3, Spring 2004, pp. 28-36.

[17] Nussbaum S and Menz G, *Object-based image analysis and treaty verification*, Springer, 2008.

[18] Lévesque J, Neville, R.A, Staenz K (2001), Preliminary Results On The Investigation Of Hyperspectral Remote Sensing For The Identification Of Uranium Mine Tailings, Proceedings ISSSR, Quebec City, 2001.

[19] Neville, R.A., Staenz, K., Levesque, J., Truong,Q.S.Bob, and Borstad, G.A. (2001) Hyperspectral Analysis of Imagery of a uranium mine site, Presentation at the ISSSR conference, Quebec City,, Canada.

[20] R. Leslie, Riggs, P and Bragin, V. (2002)Satellite Imagery for Safeguards Purposes: Utility of Panchromatic and Multispectral Imagery for Verification of Remote Uranium Mines, *Paper presented to Annual Meeting of the Institute of Nuclear Materials Management,* Orlando, Florida, 23-27 June 2002.

[21] Jasani, B., Smart, H. A., Blair, D., Stork, C. T., Snoth, J and Canty, M. (2005), Evaluation of remote sensor systems for monitoring uranium mines, Proceedings of the 27th Annual Meeting, Symposium on safeguards and nuclear materials management, ESARDA, 10-12 May 2005, London.

[22] Christopher L. Stork, Heidi A. Smartt, Dianna S. Blair, and Jody L. Smith (2006), Systematic Evaluation of Satellite Remote Sensing for Identifying Uranium Mines and Mills, Sandia Report SAND2005-7791, January 2006.

[23] Truong Q.S.Bob, Lim J, Borstad, G, Brown L.N, Parsons G, and Mulvie J. (2008), Capabilities and Potential Safeguards Applications of High Resolution Imagery, Paper Presented at ESARDA Conference, Tokyo, 2008.

[24] International Atomic Energy Agency, Uranium Extraction Technology, *Technical Report Series No. 359,* International Atomic Energy Agency Vienna 1993.

Systems Approach to Arms Control Verification

Keir Allen[1], Mona Dreicer[2], Cliff Chen[2], Irmgard Niemeyer[3], Clemens Listner[3], Gotthard Stein[4]

[1] Atomic Weapons Establishment, Aldermaston, UK
[2] Lawrence Livermore National Laboratory, Livermore, USA
[3] Forschungszentrum Jülich, Jülich, Germany
[4] Consultant, Bonn, Germany

Abstract:

Using the decades of experience of developing concepts and technologies for verifying bilateral and multilateral arms control agreements, a broad conceptual systems approach is being developed that takes into account varying levels of information and risk. The IAEA has already demonstrated the applicability of a systems approach by implementing safeguards at the state level, with acquisition path analysis as the key element. In order to test whether such an approach could also be implemented for arms control verification, an exercise was conducted in November 2014 at the European Commission Joint Research Centre, Ispra, Italy. Based on the scenario of a hypothetical treaty between two model nuclear weapons states that aims to cap their nuclear arsenals at existing levels, the goal of this exercise was to explore how to use acquisition path analysis (APA) in an arms control context. Our contribution will present the scenario, objectives and results of this exercise.

Keywords: arms control verification; systems approach; acquisition path analysis

1. Introduction

The reduction or elimination of nuclear arms is not likely to occur in the absence of a lower perceived need for nuclear weapons and high confidence that commitments are being honoured. Over more than 50 years of IAEA verification has taught us that achieving confidence requires a coherent and comprehensive picture of a state's compliance with its obligations.

The traditional IAEA verification approach was based solely on the type and quantity of nuclear materials present in a state, without regard to other factors that correlate with the actual proliferation risks of the state concerned. The State-Level Concept (SLC) was the systematic approach recently proposed as a way to increase the efficiency and effectiveness of safeguards. The SLC consists of the development of state-level safeguards approaches to identify areas of higher safeguards significance through the collection and evaluation of multi-source safeguards-relevant information to optimise future safeguards activities. By piecing together a broad range of information encompassing both information provided by the state, information derived from IAEA safeguards activities in the field and at Headquarters, and other information including open sources and third parties, it may be possible to provide state-level confidence that non-proliferation commitments are being upheld. The SLC takes into account a broader but still objective range of state-specific factors than traditional safeguards, potentially allowing greater focus on areas of higher safeguards significance.

It is important to ensure declarations made by states are both correct and complete, and this sentiment is as important for arms control verification as it is for safeguards. It is suggested that a systematic approach may prove beneficial in determining how correctness and completeness can be suitably demonstrated within the bounds of future arms control agreements. A systematic approach that assesses the military-industrial nuclear weapons enterprise of a state could also provide the holistic framework necessary for evaluating alternative verification strategies for the degree of confidence each may provide (in terms of whether a state is complying with its commitments) under the likely broad terms of such future agreements. A systematic approach to assessing arms control verification strategies could therefore help inform policymakers of the most fruitful avenues for future arms reductions or disarmament efforts.

2. Acquisition Path Analysis

The IAEA SLC methodology consists of three processes that help to develop State Level Approaches (SLA) to implementing safeguards (for more details see Cooley 2011):

1. Identification of plausible acquisition paths;
2. Specification and prioritization of state-specific technical verification objectives; and
3. Identification of safeguards measures to address the technical objectives.

Listner *et al.* (2012, 2013, 2014, 2015) demonstrated how the first of these, acquisition path analysis (APA), can be carried out using a formal methodology which is yet compatible with the principles defined by Cooley (2011).

The acquisition path analysis method uses a three-step approach:

1. The potential acquisition network is modelled, based on the IAEA's physical model and experts' evaluations

of the technical difficulty, proliferation cost, and proliferation time associated with each necessary process in the network, and estimates for the inspectorate's costs and the non-detection probabilities of the technical objectives;

2. Using this model all plausible acquisition paths are extracted automatically and ordered by to their attractiveness to a particular state; and
3. The state's and the inspectorate's options are assessed strategically.

Underlying the acquisition path analysis approach of Listner *et al.* is the application of graph theory. Table 1, below, demonstrates this.

When used to model proliferation pathways through the civil nuclear cycle and related technical capabilities of a state, the resulting model is of the form found in Figure 1.

Graph Theory	Route Planning	Acquisition Path Analysis (nuclear material)
Node	Location	Material form
Edge	Street	Process / path segment
Path	Route	Acquisition Path
Edge Weight	Attractiveness (in terms of distance, possible speed, fuel costs, eco-friendliness)	Attractiveness (in terms of technical difficulty, proliferation time, proliferation cost)

Table 1: Applying graph theory to acquisition path analysis. Each node represents a way point - a location when applied to route planning, the form of nuclear material in the case of acquisition path analysis. Each node is connected to other nodes by edges, analogous to streets, or, in APA, to processes for converting nuclear material from one form to another. Multiple edges can connect two nodes. For instance, a public footpath could be used as intended (on foot) or could be misused (travelled by car). Furthermore, a path could be a private road, i.e. one that is not accessible to the public but nevertheless offers a route to the next node. Each edge type is therefore a different but technically plausible way of reaching the next node. A route for completing a journey may include multiple streets. Likewise, an acquisition pathway within nuclear infrastructure consists of all the processes that enable states to acquire direct use nuclear material. The weight given to the path represents its attractiveness in comparison to the other routes available.

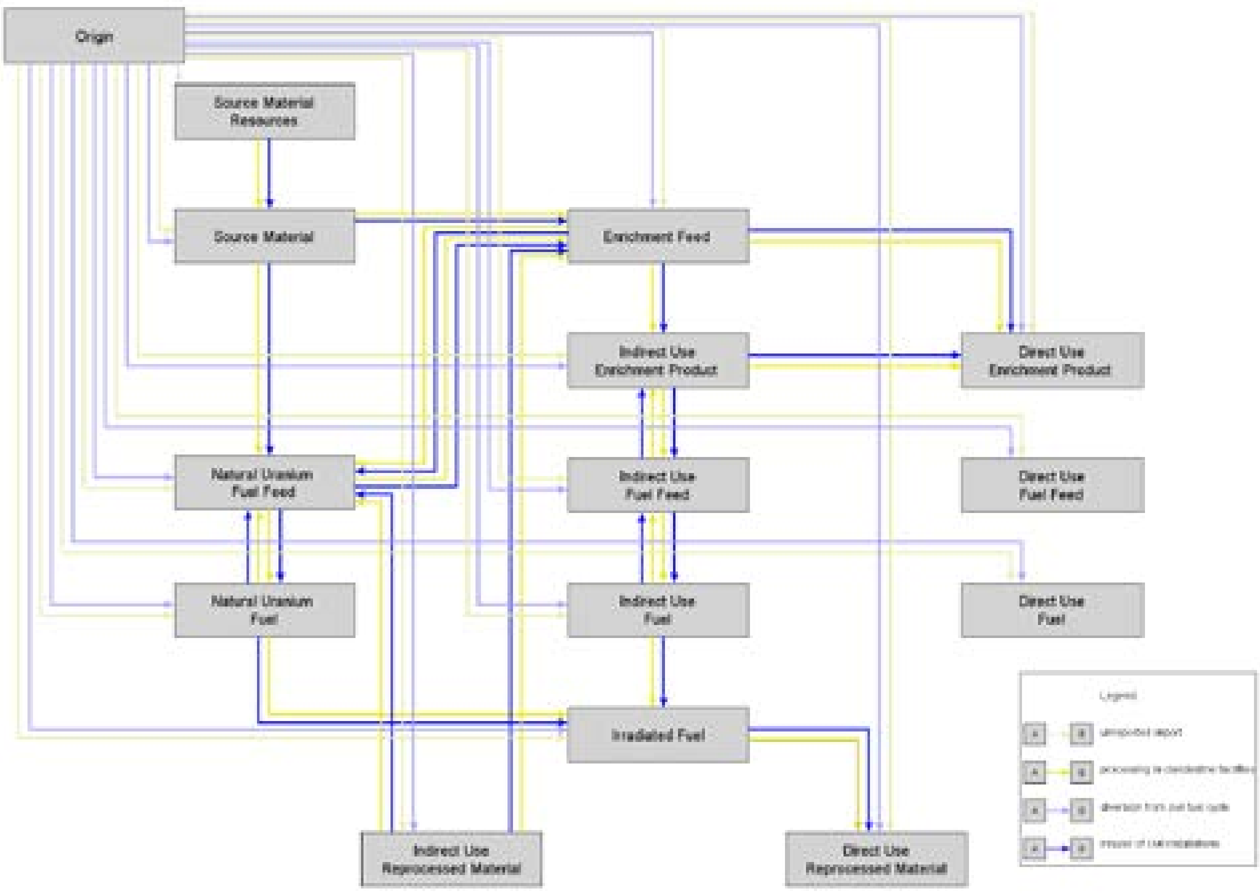


Figure 1: Civil nuclear fuel cycle and related technical capabilities of a state. Four different edge types connect the nodes, namely undeclared import of material (light yellow), processing in clandestine facilities (yellow), diversion of material from declared facilities (light blue), and misuse of declared facilities (blue). The direction of the arrows indicates the flow of material from one form to another. Pathways consist of combinations of edges and nodes that result in direct use material (i.e. material that could be used for the manufacture of nuclear explosive devices without transmutation or further enrichment).

3. Development of a systematic approach to arms control verification

What is the objective of the arms control process that we are trying to systematise? The systematic approach underpinning the SLC - at least as it applies to safeguards agreements with non-nuclear weapon states' parties to the Treaty for the Non-Proliferation of Nuclear Weapons (NPT) - is to provide credible assurance that no nuclear material required to be safeguarded is left undeclared or has been diverted for the manufacture of even one nuclear weapon or for purposes unknown. To that end, the technical objective of NPT safeguards is the timely detection of an attempt to acquire even one significant quantity of fissile material outside safeguards control, and the deterrence of such acquisition through risk of early detection.

In developing a systematic approach to verification of potential future arms control agreements, one limitation is the absence of a currently defined treaty text. The object or action that is to be regulated by the treaty needs to be defined and concepts of compliance and non-compliance need to be considered in order to provide context. Assumptions about overarching treaty objectives can be made in order to identify behaviour that might be considered non-compliant. The development of a systematic approach could then include the following three steps:

1. Modelling of a cheating network and identification of cheating pathways. This is an assessment of the potential attractiveness of the alternative routes available to a state wishing to cheat on its treaty commitments;

2. Determination of technical verification objectives, including identifying the required detection probabilities for each area of a potential cheating network. It is assumed that high confidence verification would call for high detection probabilities for areas of highest risk; and

3. Identification of the technical and administrative measures that would provide the required detection probabilities. This would be expanded beyond classical inspections and could include all types of measures related to the field of interest.

3.1 Application of APA to the production of fissile material in states possessing nuclear weapons

The use of acquisition path analysis to assess pathways to obtain direct use nuclear material in states possessing nuclear weapons is relatively simple. For the purposes of developing the methodology, we considered the verification of nuclear materials subject to international commitments, such as existing VOAs, INFCIRC/66 agreements, or under a future FMCT, in a state possessing nuclear weapons.

Since nuclear materials are the focus of regulation for the agreements considered in this section[1], possible non-compliant behaviour could include the misuse of civilian facilities or diversion of materials from civilian facilities (which are a subset of the four types of non-compliant behaviours included in the APA model for NPT NNWS safeguards in Figure 1), or diversion of materials already located within military fuel cycle to uses prohibited by the agreement, or misuse of military facilities in ways prohibited by the agreement. The additional cheating pathways need to be added into the model of the state in the form of additional edge types. These are illustrated for a simple example in Figure 2.

We considered the situation presented by two different types of states in order to see the effects on the relative attractiveness of different pathways under three scenarios. The states considered are:

1. A Nuclear Weapons State (NWS) within the NPT and with a Voluntary Offer Agreement (VOA). A state having signed a VOA must not use the facilities under this agreement to produce material for use in weapons.

2. A nuclear weapons possessing state outside NPT, but with INFCIRC/66 in-force (facility or item-specific commitments in force). A state outside the NPT but with INFCIRC/66 type agreements must not use these facilities or items for military purposes.

The three example scenarios are listed below. Each scenario is based on the extent of the infrastructure present in the country and the extent of the commitment made by it. Included is a comment regarding the construction of the acquisition path model, including the relative attractiveness of different pathways (and therefore possible courses of action for the verification body).

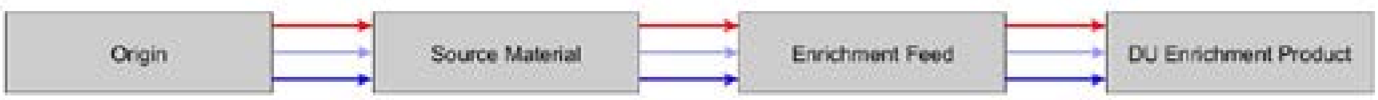


Figure 2: Exemplary cheating paths. Each node represents a form of material. The direction of flow of material indicates the usefulness of the material for weapons purposes, from least useful to most useful (with the most useful being Direct Use, or DU) material. The edge types included represent diversion (light blue) and misuse (bright blue) in the civil nuclear fuel cycle, and red arrows which consider the military fuel cycle: Diversion from the military fuel cycle is represented by the red arrow from "origin" to "source material" and military processing is represented by all the remaining red arrows.

[1] Section 3.2 below extends the discussion to agreements that would restrict nuclear weapons and associated production infrastructure

1. *A state with a complete military fuel cycle (with no monitoring) and with some or all civilian facilities under safeguards*

   In this case, the safeguards are applied to civilian facilities under INFCIRC/66 or a VOA, but a comprehensive agreement, (e.g. INFCIRC/153) is missing and so the production of fissionable material in undeclared facilities would not be violating any agreement. Military facilities are also exempt from inspections. When an activity is allowed, it can be represented in the model but the detection probability (for calculating attractiveness) is set at 0% because an allowed activity will not need to be monitored. In this situation, military or undeclared pathways will remain the most attractive pathways for producing materials for weapons. It is assumed that there will be no need for misuse or diversion from the declared civil fuel cycle. The risk of sanctions, if non-compliant behaviour is detected (such as pursuing a pathway that uses civil installations under international surveillance) may be sufficient to deter misuse and diversion within civilian facilities.

   If a multilateral treaty was in force that prohibited undeclared activities, then detection methods might be employed that would cause a rise in the detection probability for clandestine operations, leading to a decrease in the attractiveness of pathways that use clandestine facilities. Military pathways would remain unaffected.

2. *A state with an incomplete military fuel cycle and with civilian facilities under safeguards*

   In order to model the existence of gaps in the military fuel cycle in the APA model, they would be represented by the absence of certain diversion edges in the network or a change in the attractiveness of pathways that require the gap to be bridged through either the development of new military fuel cycle facilities or recourse to misuse of any relevant safeguarded civilian facilities. Where gaps in the military cycle are present, a state could find pathways which include misuse of civilian facilities much more attractive. Effective verification of these pathways could deter non-compliant behaviour. Appropriate monitoring measures (possibly including increased monitoring in particular facilities) would increase the risk to the state of detection. If the risk (and costs) of detection are high, the state should be deterred from non-compliant actions.

3. *A state in which military facilities and materials are placed under a fissile material control regime*

   If military facilities and materials are put under a multilateral treaty to control the production of all fissile materials, then installations that produce and use defined fissile materials, whether civilian or military, would presumably be subject to monitoring and verification. Therefore pathways that use military facilities as a source of materials for nuclear weapons could be treated in much the same way as any other pathway for analysis purposes. To deter the use of these paths in violation of treaty commitments would require a significantly increased monitoring/inspection effort because of the additional facilities and material stocks that need monitoring. The ability to verify a baseline declaration and knowledge of past production could be a key factor if past production were to be included under the treaty.

The above discussion demonstrates how acquisition path analysis could be expanded to assess material acquisition pathways in states with nuclear weapons, but in order to develop a systematic approach to arms control or disarmament verification, the methodology will need to address the links between nuclear material production infrastructure (whether civilian or military) and nuclear weapons themselves, and the infrastructure used to support weapon production and deployment.

3.2 Application of APA to nuclear weapons and nuclear weapons military and industrial infrastructure

To further the development of a systematic approach to arms control, work is needed to expand the physical model to include the nuclear weapons enterprise of a state. The aim is to produce the inter-connecting network of nodes and edges that maps the flows of nuclear materials and weapons throughout a state.

As stated previously, the definition of a cheating pathway depends on the terms and objectives of the treaty being assessed. By assuming overarching treaty objectives, cheating pathways can be mapped and the relative attractiveness or usefulness of a particular cheating pathway (from the perspective of the cheating state) can then be considered.[2] This process may help arms control practitioners to formulate suitable verification objectives for all cheating pathways.

The task of formulating verification objectives for future treaties is complicated by the fact that many of the processes, actions and infrastructure that might indicate non-compliance in an NPT non-nuclear weapon state may be present as a matter of course in a weapons state. For treaties that do not altogether prohibit nuclear weapons, the existence of weaponisation indicators alone will not be sufficient for determining non-compliance. Verification measures must therefore be capable of discriminating between permitted, declared weapons related activities and non-compliant activities. Even if the total elimination of weapons is the goal, there is likely to be a significant draw down period in which certain actions are sanctioned.

In order to sufficiently model a future in which all weapons design and manufacture has been prohibited, cheating

[2] It is recognised that expert judgement will be required at this stage.

pathways that exploit the reconstruction of existing warhead designs (which have previously been proven through testing) should be clearly distinguished from pathways that require the development of novel weapons (which may require an element of testing in suitable facilities). The indicators which may be used to monitor these alternative pathways may well be significantly different from one another.

A further consideration for the development of a systematic approach to arms control verification is to ensure that the definition of a treaty accountable item (e.g. a weapon) is suitably constructed such that that any item which could lead to accusations of non-compliance if not declared (and later found) can be suitably investigated and proven to be either complaint or not.

Furthermore, a systematic approach to arms control verification should result in the identification of verification objectives that take account of the challenges presented by the potential (un)availability of information pertaining to nuclear weapons programmes and the supporting industrial and military infrastructure. Commitments under Articles I and II of the Nuclear Non-Proliferation Treaty (NPT) will impede the ability of NWS to disclose to NNWS categories of information that may be considered proliferation-sensitive. Equally NWS may be unwilling to share details of their arsenals, operational factors or infrastructure with any other state, nuclear weapons possessors or not, because of the potential impact of disclosure on the effectiveness of extant military strategies and national security postures. It may prove necessary to develop generic models of infrastructure in order to protect sensitive information.

4. Workshops

A workshop held at the European Commission Joint Research Centre in Ispra, in conjunction with the 2014 Fall Meeting of the ESARDA Verification Methods and Technologies (VTM) Working Group (WG), began to explore the application of a systems approach and acquisition path analysis to arms control verification.

An exercise scenario was presented to the working group that bridged the gap between safeguarding of materials and New START style accounting of warheads through verification of nuclear weapon delivery systems. Under the exercise scenario, a treaty had been signed by two states that required each state to cap the total number of

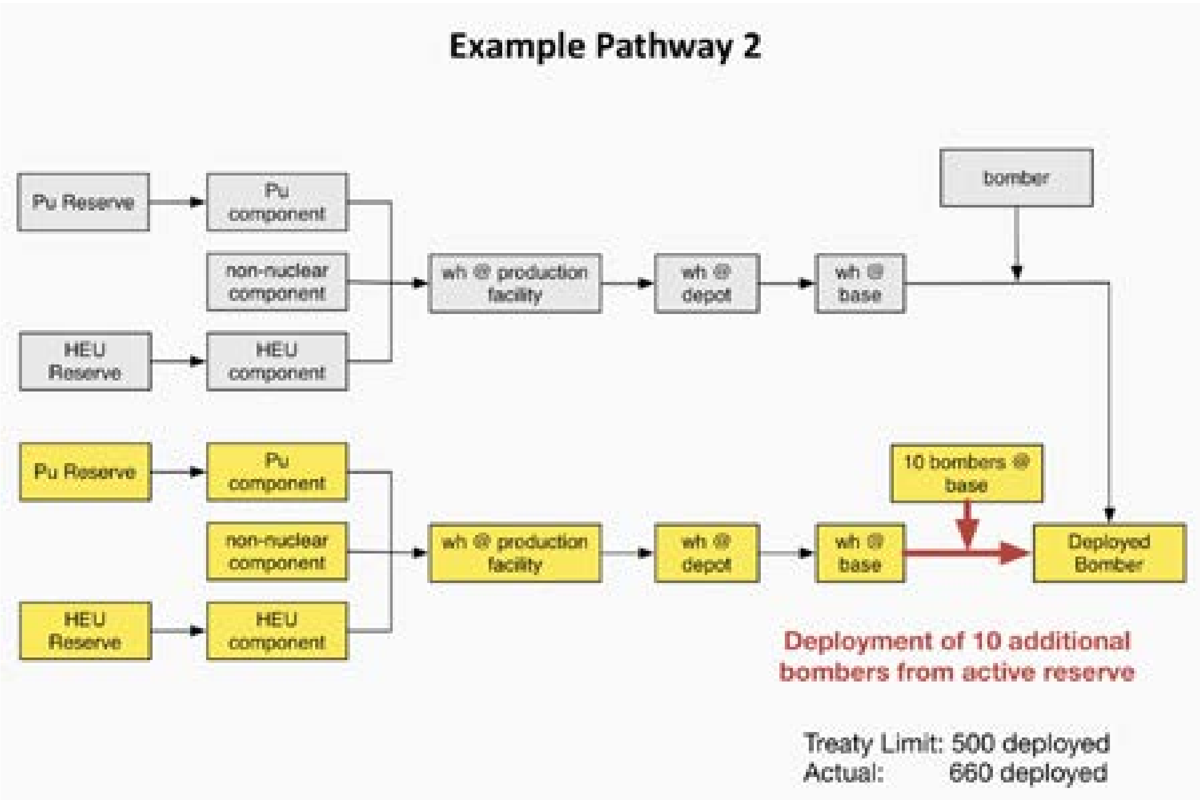


Figure 3: An example pathway discussed during the workshop. The weapons enterprise is represented as a linear flow from weapons useable material to deployed weapons. Declared facilities are coloured yellow, undeclared facilities are coloured grey. In this example, the pathway exploited is indicated by bold, red arrows. The pathway shows the deployment of warheads from the declared reserve stockpile onto reserved aeroplanes. The action increases the total number of deployed warheads from 500 to 660, significantly breaching treaty limits. Neither the existence of the declared reserve aeroplanes, nor the declared stockpiled warheads, would breach the treaty so long as they remain in reserve and are not mated. More complex pathways exist which may involve combinations of declared and undeclared materials, components, weapons, facilities, process, stockpiles and delivery vehicles.

warheads in its nuclear arsenal and to cap the number warheads it deployed to no more than 500. Each state maintained a full nuclear weapons enterprise from fissile material production and weapons design, through manufacture and deployment on multiple delivery systems. Members of the working group were asked to discuss potential pathways that one of the states could take to cheat on its treaty commitments and then to consider how to apply acquisition pathway analysis and a state level approach to the scenario. Figure 3, below, shows one of the illustrations used during the workshop. It shows the weapons enterprise and a simple example of cheating. The slides were developed to help facilitate discussions and were not developed as mathematical graphs.

The workshop results are introduced below:

4.1 Verification objectives

Under the exercise scenario, since each side is permitted to maintain a deterrent of 500 deployed weapons, strategic stability may not be affected by small fluctuations around this number. With this in mind, what is a suitable objective for the verification system and what level of activity should the system be capable of detecting?

A number of terms need to be defined:

1. What constitutes a significant breach under the treaty?
2. What level of accounting accuracy is required in order to detect cheating before a significant breach occurs?
3. What constitutes timeliness for detection?

If strategic stability is tolerant of afew additional warheads, then a material breach of commitments may only be considered at higher levels of noncompliance. Nontheless, a truly significant breach could affect strategic stability. It follows that the verification system must be capable of detecting even low levels of cheating since such behaviour may be indicative of larger scale cheating, and it must do so well before a significant breach occurs.

It should be noted that the level of cheating that constitutes a significant breach may differ between partners.

The definition of a significant breach will impact upon the requirements for accuracy and timeliness. Meanwhile, the impact of accuracy and timeliness requirements on resource requirements will also need to be considered.

It might be that a suitable definition of a significant breach and the requirements for accuracy and timeliness for treaties where large numbers of warheads remain deployed are not suitable when deployed numbers are considerably reduced. Definitions that are tolerant of discrepancies early on when permitted inventories are high, may cause uncertainties to propagate through the accounting and verification process until they impact upon treaties that stipulate lower numbers of weapons. It is therefore important to develop a mechanism for achieving consensus on verification objectives.

4.2 Making declarations that are verifiable

It is important to define the controlled items in such a way that declarations made about them in relation to the treaty can be verified effectively. Because of the secrecy surrounding weapons systems, verifying the correctness of a declaration (e.g. Verifying that an item declared to be a weapon is a weapon) could be challenging. During the workshop a discussion was held concerning the necessity of verifying correctness under the scenario being discussed, since any items declared to be weapons would still count towards the 500 limit of the state making the declaration. Declared items would be accepted as warheads with minimal verification and the emphasis of the inspection regime would then be to verify the completeness of the declaration (i.e. to ensure the absence of undeclared items that could be warheads). This approach was judged to be reasonable since even if declared items where meticulously verified to be warheads, the inspection process would still need to provide adequate assurance of the absence of further undeclared items that could be warheads.

For this scenario, the treaty objective is to limit deployed and non-deployed warheads. The correctness of the declaration should therefore be measured against the status of deployment (e.g. deployed or non-deployed, in reserve, inactive, disassembled, or weapon component). Following this convention, the declarations were framed in a way that could also allow items declared as none of those categories to be verified as ‘none of the above’. Significant detail concerning the activities of the nuclear enterprise and locations of declared weapons may need to be shared, and updates exchanged regularly, to facilitate verification of deployment status.

An extension of the fissile material acquisition path analysis approach to include nuclear weapon lifecycle steps relevant to the monitoring of nuclear weapons treaties could include the application of graph theory using appropriate new types of nodes and edges as suggested in the right hand column of Table 1.

Graph Theory	Route Planning	Acquisition Path Analysis (materials)	Weapons Verification
Node	Location	Material form	Weapon status/form
Edge	Street	Process / path segment	Deployment process/path segment
Path	Route	Acquisition path	Deployment path
Edge Weight	Attractiveness	Attractiveness	Attractiveness

Table 2: Use of graph theory to analyse acquisition paths relevant to nuclear weapon treaties, where nodes would correspond to deployment status.

4.3 Measures of attractiveness

In order to rank pathways according to attractiveness we may need to bear in mind that a state might cheat on its commitments in order to achieve one of a variety of strategic goals or priorities. Factors that contribute to attractiveness will need to be weighted appropriately in order to account for this.

The most attractive pathways could be different if the objective is to expand the size of the national stockpile or to increase the degree of technical sophistication of the stockpile (which might or might not be considered cheating, depending on the particulars of the treaty). Some potential cheating pathways include secretly excluding weapons from baseline declarations, diverting materials or components during dismantlement, and the undeclared production of weapons. The link between monitored nuclear material (either civilian or military) and weapon production will need to be considered to achieve confidence that new production is not occurring.

The working group discussed suitable metrics for identifying the pathways most likely to be exploited under this scenario. Comparison was made with the six proliferation resistance metrics defined in the Evaluation Methodology for Proliferation Resistance &Physical Protection of Generation IV Nuclear Energy Systems (2011). Broadly, the metrics (or, more accuarately, their analogues in the nuclear weaponisation context rather than fissile material acquisition context) were considered to be suitable. The metrics are titled below as per the Gen IV definitions for simplicity and are accompanied by a summary of the discussion of each:

- *Proliferation Technical Difficulty* – Since weapons' states already maintain the full weapons enterprise, it is assumed they maintain the ability to deploy existing weapons, build additional weapons or modify stockpiles, and so a technical barrier to exploiting a pathway may not exist. Furthermore, such actions could be masked to a large extent by allowed processes. Nevertheless, some pathways may require mobilisation and coordination of greater resources than other pathways (e.g. the deployment of reserve warheads onto reserve bombers may be accomplished more simply than building a clandestine stockpile of new weapons and secretly loading them onto a submarine). *Proliferation Difficulty* and *Detection Probability* were discussed in terms of the stealth required to successfully exploit a pathway.
- *Proliferation Cost* – The unequal cost of certain pathways relative to others is evident (see example in previous bullet point). Nevertheless, capital and operational costs of pathways may already be accounted for in national budgets. Only pathways requiring significant capital investment may be deemed less attractive to a state wishing to cheat. Overall, cost may not be a primary factor in the decision to exploit a pathway.
- *Proliferation Time* – The minimum time required to deploy a strategically significant quantity of additional weapons. The significance of the proliferation time is closely tied to the strategic goals of the state. For example, a short proliferation time allowing the deployment of large numbers of weapons very quickly might be considered strategically advantageous in some situations by a cheating state. Equally, a long proliferation time associated with a very stealthy build up of a clandestine stockpile might be considered attractive in other circumstances. Since both routes could be attractive, this should be appropriately reflected in the weighting applied for calculating attractiveness.
- *Fissile Material Type* – Not discussed in great detail since stockpiles of material were declared under the treaty.
- *Detection Probability* – With no standard set of verification measures assumed to have been agreed or allowed under this treaty, cumulative detection probabilities cannot be determined in advance. This reflects well the situation for future nuclear warhead arms control verification where no specific verification technologies or methods have been agreed. In this case acquisition path analysis can be used to identify where specific verification measures might be of most benefit. Design requirements for technologies can then be stipulated based upon the identified verification requirement and location specific regulations. A systems level analysis can therefore help identify technology requirements for future treaties. Weapons or weapons components require sufficiently robust monitoring to ensure that their location remains known to the inspectorate to a high level of confidence, thus ensuring that they are not deployed in breach of treaty commitments. High confidence might also be required in ensuring that only declared items can interact with declared delivery systems. Detection probabilities for undeclared items at declared deployment sites must

therefore be sufficiently high and the detection times in such instances should also be very rapid.

- *Detection Resource Efficiency* – The efficiency in the use of staffing, equipment, and funding to apply verification measures across different parts of the weapons enterprise. There may be points in the nuclear weapons enterprise where the ability to verify declarations with high confidence would be particularly beneficial. In the case of the exercise scenario, the monitoring to ensure delivery systems only carried the declared number of warheads was identified to be important. Nonetheless, the verification focal point in other scenarios could shift depending upon the aims and objectives of the treaty.

5. Benefits of a systems level analysis

A system level analysis can help identify verification requirements for arms control based upon the strategic goals of treaty. It can help identify useful ways of framing and defining verification objects, processes and timescales and can help identify the types of technologies needed to provide monitoring capabilities in specific locations, which means technology requirements can be stipulated having taken account of facility or location specific restrictions. With regard to technology development, defining the combination of purpose and location is important because solutions can then be tailored to perform the defined functions in a way that fully meets the potentially unique safety, security and operational restraints of the specific location.

Analysis at the systems level ensures that compliance verification requirements are not skewed to any single point in the overall regime. Instead specific verification objectives are defined at each point in a process for all points in time, and suitable technical measures can be identified to fulfil those objectives. By taking the systematic approach, all the ways a state could breach its commitments are assessed in the context of the strategic objectives of the treaty and a suitable amount of resource can be allocated to ensure all risks are addressed appropriately. A state-level methodology could therefore help inform the direction of future negotiations, help direct present day technology R&D efforts, and provide a framework for the assessment of possible verification regimes.

The need to make assumptions about the overarching objectives of future treaties means the resulting models are unlikely to be entirely accurate. Nevertheless, models can provide insight into the most pressing requirements if they are broadly correct in their assumptions. In that case they can be used to inform arms controllers of likely challenges and the requirements any candidate solutions may need to satisfy, and they can be used by states to prepare for any commitments assumed under a similar treaty.

It is likely that a systematic or state level analysis is already performed by states (such as US and Russia) that have extensive experience in arms control agreements. In the context of bilateral arms control agreements, weapons possessing states may have a basic understanding of a nuclear weapons complex, in particular the competing needs for effective verification, protection of national security information, and upholding NPT Article VI commitments (in the case of the NWS under the NPT). The systems approach can be used under such circumstances to assess different routes to achieving overarching treaty objectives. Systems level analysis can provide clear verification objectives for site visits, based upon information already provided by the inspected state. With clear verification objectives, managed access procedures can be defined that meet those objectives whilst protecting sensitive information.

The approach may help those weapons possessing states with significant nuclear arms control experience to assess more broadly the various ways of providing the information deemed necessary for compliance verification, whilst ensuring the continued protection of information they consider sensitive. This could include highlighting where the development of new operational tactics could offset any perceived vulnerability arising from a loss of secrecy.

While a formal approach may add value to the analysis performed by those experienced weapons possessing states, a major benefit may be the common framework it can provide to states without the capacity for or experience with analysing arms control verification regimes. In this case, the state level approach can promote understanding about the strategic and technical challenges associated with arms control verification. For weapons possessing states with little nuclear arms control experience and for non-nuclear weapon states, this formal approach to assessing arms control regime requirements and potential solutions could be very useful.

6. Further considerations

The development of a state-level approach to modelling acquisition pathways is more advanced for international safeguards on materials in civilian programs than for the verification of treaties limiting nuclear weapons, but work can be done to further expand the models and make the linkages between material and weapons cycles. As mentioned earlier, a challenge to the successful modelling of a state's nuclear weapons complex arises from the fact that many of the processes, actions and infrastructure that might constitute an indicator of non-compliance in an NPT non-nuclear weapons state may be present as a matter of course in a weapons state. True indicators of non-compliance with agreements may therefore be much more subtle in nuclear weapons states, requiring detailed information on the level of expected activity in the weapons facilities of the state. Verification methods need to be fine-tuned such that sanctioned activities do not mask cheating.

Furthermore, non-compliant actions with potentially significant consequences (i.e. a material breach) could take place on very short time scales, and so detection times must be commensurately short. Therefore the level of intrusiveness required to effectively monitor sanctioned activities may be considerably greater than for current or historical agreements. All of these factors can be incorporated into suitable systems level models to improve the analysis of links between materials and weapons.

The challenges associated with the protection of national security and proliferation sensitive information must be taken into account as a realistic physical model is developed that incorporates further intrusiveness. Existing ideas for managing access for routine and challenge inspections or new ideas will need to be considered. Verifying that declared items are situated in their declared location may prove to be a relatively straightforward matter of accounting, assuming suitable managed access procedures can be developed. In contrast, verifying the absence of undeclared items, either in declared facilities or undeclared facilities, could be a far more challenging task. Nevertheless, as suggested by the results of the workshop, early priorities in this area could focus on ensuring that undeclared items cannot successfully be mated with delivery systems and there are parallels to this approach in how verification of the absence of warheads on delivery systems is accomplished under New START at present.

Any advancement in arms reductions and disarmament is likely to proceed on a step-by-step basis. Bilateral agreements are likely to provide the steps that will pave the way for more multilateral implementation. For example, future US/Russia arms control treaties limiting warhead numbers may develop the blueprint for facility monitoring and inspection activities, whilst transparency and confidence-building measures implemented amongst the de facto nuclear weapons states could provide the foundation for monitoring activities along the lines of those developed by Russia and the US, to take place at a later date.

Acknowledgements

The co-authors would like to thank all the participants of the workshops held at the EC JRC ITU Ispra (November 2014) and at Lawrence Livermore National Laboratory (July 2015) for their interest, fruitful discussions and inputs, and George Anzelon, Lawrence Livermore National Laboratory, for his valuable comments on the paper.

References

[1] J.N. Cooley. 'Progress in Evolving the State-level Concept' *Seventh INMM/ESARDA Joint Workshop Future Directions for Nuclear Safeguards and Verification, 2011*

[2] Listner, C.; Canty, M.; Rezniczek, A.; Stein, G. & Niemeyer, I. 'A Concept for Handling Acquisition Path Analysis in the Framework of IAEA's State-level Approach' *Proceedings of the INMM Annual Meeting, 2012*

[3] Listner, C.; Canty, M.; Rezniczek, A.; Stein, G. & Niemeyer, I. 'Approaching acquisition path analysis formally - a comparison between AP and non-AP States' *Proceedings of the 35th ESARDA Annual Meeting, 2013*

[4] Listner, C.; Canty, M.; Rezniczek, A.; Stein, G. & Niemeyer, I. 'Evolution of Safeguards - What can formal Acquisition Path Analysis contribute?' *Proceedings of the INMM Annual Meeting, 2014*

[5] Listner, C.; Niemeyer, I.;Canty, M.;Murphy, C.L.; Stein, G. & Rezniczek, A. 'Acquisition Path Analysis Quantified – Shaping the Success of the IAEA's State-level Concept' *Journal of Nuclear Materials Management* 43(4), 2015, pp. 49-59

[6] Nuclear Threat Initiative (NTI) 'Innovating Verification: New Tools & New Actors to Reduce Nuclear Risks.' *Verifying Baseline Declarations of Nuclear Warheads and Materials, 2014*

[7] The Proliferation Resistance and Physical Protection Evaluation Methodology Working Group of the Generation IV International Forum, '*Evaluation Methodology for the Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy System Revision 6*', Sept 2011, ref: GIF/PRPPWG/2011/003, https://www.gen-4.org/gif/upload/docs/application/pdf/2013-09/gif_prppem_rev6_final.pdf,accessed 12/05/2015

LLNL-JRNL-679309

Safeguards-relevant information collection from small holders - experiences and challenges

Erika Sundén, Martina Dufva, Joakim Dahlberg

Nuclear Non-proliferation and Transport
Swedish Radiation Safety Authority
Solna Strandväg 96, SE-171 16 Stockholm, Sweden
E-mail: erika.sunden@ssm.se

Abstract:

A number of universities, research institutions, hospitals, and other businesses are in possession of relatively small amounts of nuclear material. In some cases the material is in activities related to the nuclear fuel cycle, but there is also a wide variety of other applications. Regardless of application, material accountancy must be reported to the European Commission (EC) and, in Sweden, to the Swedish Radiation Safety Authority (SSM). However, checking the completeness and correctness of the reports from operators with very small amounts of nuclear material can easily be forgotten or viewed as being less important. Starting in the beginning of 2013 SSM increased its effort in this area and began working more actively gathering information and checking its correctness. Informing the operators in possession of nuclear material of the rules and regulations is a major part of this work, as it has been noted that the knowledge level of safeguarding nuclear material in many locations is very low.

This paper will give a description of the work being performed by SSM to ensure that information related to the possession of nuclear material are gathered and correctly declared. It will give an overview of the different procedures that are applied to different categories of small holders in Sweden (where the differences are mostly due to historical reasons). It will also entail some of the challenges met along the way; such as explaining to radiographers that for nuclear non-proliferation purposes it is the shielding uranium container which is of interest, not the isotope emitting the radiation. What we have experienced being the major differences between collecting information from small holders as compared to larger nuclear facilities will also be outlined. The paper ends with an outline for future work.

Keywords: small holders, information collection, LOF

1. Introduction

Sweden is a country with a long tradition of nuclear related activities. Already in the 1940s both civil and military nuclear programmes were developing. However, signing and ratifying the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) [1] in 1970 officially set an end to the military dimension of the Swedish nuclear programme. In the 1970s and 80s the civil program grew and industry related to the civil nuclear fuel cycle expanded [2]. Currently Sweden has ten light water reactors in operation and two permanently shut down. There is also a fuel fabrication factory and a research facility that up until 2005 contained a research reactor in operation. For the back-end of the fuel cycle a central interim storage facility for spent nuclear fuel was built and started operation in 1985. A final geological repository is planned and a first round of applications from the Swedish Nuclear Fuel and Waste Management Co (SKB) are currently under review by SSM and the Land and Environmental Court.

In addition to the large fuel cycle related facilities in Sweden there are also a number of small holders of nuclear material. Some of the nuclear material, especially at schools and universities, was purchased a long time ago and predates the Swedish signing of the NPT. The amounts of material and applications vary and the holders can be anywhere from a small company with a handful of employees to large research institutions with hundreds of people within the organization. The number of small holders is not static over time, as a contrast to the very long-term operations of larger nuclear fuel cycle facilities, new holders can emerge quickly and others disappear by selling or transferring their material elsewhere. These small holders are subjected to requirements for safeguarding their material and should follow the same rules and regulations in this area as the power plants and the other large facilities. This paper will describe the work being done by the unit on Nuclear Non-proliferation and Transport at the Swedish Radiation Safety Authority (SSM) for ensuring the completeness and correctness of the declarations submitted from this section of nuclear material holders.

2. Overview of holders of small amounts of nuclear material in Sweden

SSM keeps a national registry over all known nuclear material in Sweden. This registry includes both the nuclear fuel cycle material as well as the nuclear material at each small installation. When nuclear material is discovered and registered, the national registry is updated.

As of April 2015 there are in total 21 registered holders of small amounts of nuclear material in Sweden. All together they are in possession of approximately 0.6 kg low enriched uranium, 1,000 kg natural uranium, 1,300 kg depleted uranium and 11 kg of thorium. The bulk part of the depleted uranium is in the form of radiation shielding devices.

The sum of the holders´ combined amount of highly enriched uranium and plutonium is of the order of 50 g in each category. The holders approximate locations can be viewed in figure 1. “WSWE” is the Material Balance Area (MBA) code for the Swedish national Location Outside Facility (LOF). “CAM” refers to the holders which are organized within the European “Catch-all” MBA. “Own MBA” refers to holders which have their own MBA-code. More on the differences between these three categories of nuclear material holders will be explained in section 3.2.

The registered holders of small amounts of nuclear material include radiographers, scrap metal yards, recycling facilities, universities, laboratories, research institutions and hospitals. They have different types of nuclear material for different applications. SSM suspects that there are more installations in possession of nuclear material than presently registered and work is on-going to investigate this. As of yet, a structured approach to such an investigation has not been employed, but there has been some activities to this end. One out-reach activity aimed at radiation experts at universities has been conducted and plans exists for similar out-reach activities to other professional groups, such as radiographers. Information about unregistered holders can come from the installation itself or through an already registered holder, from open sources, or from other units within SSM, e.g. Occupational Practices and Work Activities. When information about unregistered holders reaches SSM, the installation is usually included in WSWE and the material stock is reported to the European Commission.

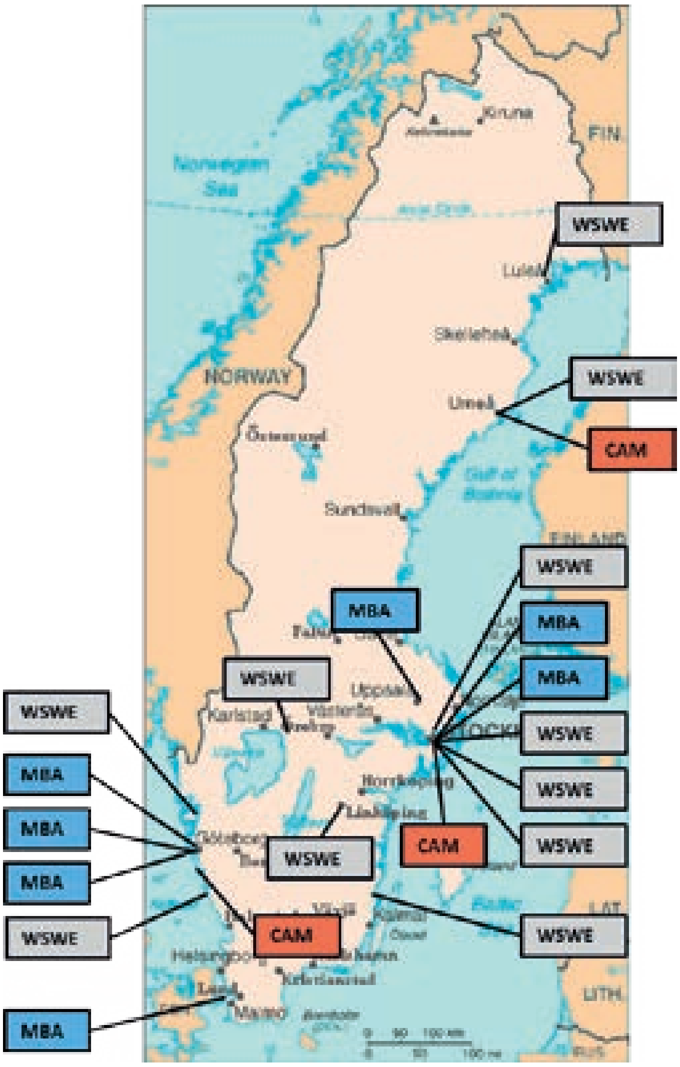


Figure 1: A map of Sweden with the general location of registered holders of small amounts of nuclear material. The labels refer to the three categories of holders. “WSWE” is the national LOF, “Own MBA” means holders which have separate MBA-codes, and “CAM” refers to the holders included in the European Catch-All MBA. The categories are more thoroughly explained in section 3.2. The map reflects the situation as of April 2015 (map adapted from WikiMedia Commons).

When an installation is found in possession of nuclear material which has not yet been reported, the reason usually given by the holder is lack of knowledge. The rules and regulations applicable to a holder of nuclear material is then provided and explained and the holder tries to provide the correct information to SSM. This process is sometimes quite slow because there is often a need to search for documentation and/or make measurements of the material to determine its composition.

3. The legal framework for Swedish nuclear non-proliferation

Sweden is a party to the NPT and a member of the European Union (EU). Following that, Sweden has accepted safeguards by the International Atomic Energy Agency (IAEA) to control that all nuclear material declared by the state is not misused. The control is based on a safeguards agreement concluded with the state. Before entering the EU a Comprehensive Safeguards Agreement (CSA, INFCIRC/234 [3]) according to the model agreement INFCIRC/153 [4] was in force in Sweden. By joining the EU the agreement was replaced with INFCIRC/193 [5] where the major difference lies in that the European Commission (through Euratom) is the contact for all Swedish facilities. In 2000 Sweden ratified the Additional Protocol (AP) [6] and in 2004 the AP entered into force. Sweden decided to be a so called ‘non-side-letter state’, i.e. Sweden is responsible for Articles 2a(i), 2a(iv), 2a(ix), 2a(x), and 2b(i), which entails declaring nuclear fuel cycle-related research, the manufacturing of certain products, the export outside the Community of equipment and non-nuclear material listed in Annex II of the AP, and general plans for the succeeding ten-year period relevant to the development of the nuclear fuel cycle. For Articles 2a(iii) and 2a(viii), which entail providing the IAEA with a general description of each site and information about processing of intermediate or high-level waste, there is a shared responsibility between Sweden and the European Commission. Sweden decided to nominate its Competent Authority, the Swedish Radiation Safety Authority, as the site representative for each site in

Sweden. For articles 2a(v), 2a(vi) and 2a(vii), which entail declaring information about mines and source material, and quantities of material exempted from safeguards under article 36 or 37 under INFCIRC/193, the European Commission is the responsible party.

As a member of the European Union, the Commission Regulation (Euratom) no 302/2005 [7] has the same legal standing as a national law in Sweden. To fulfil the articles in the Additional Protocol the sentiment of the articles has been incorporated into Swedish national law. This was done by updating the Act [8] and Ordinance [9] on Nuclear Activities, the Act [10] and Ordinance [11] on Inspections according to International Agreements on the Non-proliferation of Nuclear Weapons, and in regulations prescribed by SSM (mainly SSMFS 2008:3 [12]).

There are thus EU-regulations and national laws and regulations that implement the international agreements on safeguards (such as the AP). In addition to this there are also other obligations stemming from national laws and regulations. In the Act on Nuclear Activities it is specified that nuclear activities requires a licence, either from the Government or from SSM. The Ordinance on Nuclear Activities specifies the limits of the amount of nuclear material an entity can hold before it needs a licence from the Government or from SSM, and what amounts only need to be reported although does not require licence. The Act on Nuclear Activities gives SSM the right to issue the regulations necessary to ensure compliance with obligations in agreements aimed at preventing the proliferation of nuclear weapons and unauthorised dealings with nuclear material including spent nuclear fuel.

3.1 National rules and regulations for small holders

As mentioned in the introduction, the holders of small amounts of nuclear material must follow the same rules and regulations for safeguarding their material as the large nuclear fuel cycle related facilities. Of course, not all provisions are applicable to all types of activities or businesses, but the same basic rules apply. In particular, all holders must have control over their stock of nuclear material. This entails keeping a current list over all nuclear material and to up-date it as needed.

Most of the additional rules regarding nuclear non-proliferation for holders of nuclear material are specified in regulations prescribed by SSM, where the majority can be found in SSMFS 2008:3 (The Swedish Radiation Safety Authority's Regulations on the Control of Nuclear Material etc.) [12]. The main national rules on nuclear non-proliferation applicable to the holders of small amounts of nuclear material are summarized here:

All holders must have control over their stock of nuclear material. All changes to the inventory must be reported to SSM within three business days. The nuclear material must be stored so that identification and verification can be made, and at an inspection there must be personnel with enough competence present so that inspectors can fulfil their inspection tasks. After a request by SSM, the holder must establish a site description over the buildings or parts of buildings where nuclear material is used or stored. All holders must also appoint a person responsible for safeguards within their organisation and report a point of contact for communications with the Authorities (the State Authority, the European Commission, and the IAEA). All holders must have a system and an organization with enough financial and personnel resources to ensure these tasks are fulfilled.

Holders of nuclear material are subject to additional obligations besides those aimed at ensuring nuclear non-proliferation summarized. For example, they must have a system for physical protection of the material, a waste management system, financial means to take responsibility for the material until it is placed in a final repository, and handle the material correctly to limit the radiation exposure to those coming in contact with the nuclear material. Some of these additional rules derive from the Radiation Protection Act [13] which aims at protecting humans, animals, and the environment from harmful exposure to radiation. An amount of nuclear material can be exempted from license according to the Ordinance on Nuclear Activities but still require a license according to the Radiation Protection Act.

3.2 Different categories of small holders and international regulations

In Sweden the holders of small amounts of nuclear material are divided into three categories, here called: “Own MBA”, “National LOF”, and “Catch-All MBA”. With different combinations of exemptions and derogations in these categories the set of international rules they must abide by will vary between them, in addition to the national rules and regulations described in the previous section.

“Own MBA” refers to the holders which have their own separate Material Balance Area (MBA) code. In general, they report their inventory and changes to it directly to the European Commission using the Euratom Nuclear Materials Accounting System (ENMAS) reporting tool. They are also obligated to provide annual activity reports, Basic Technical Characteristics (BTC), site descriptions, and keep them updated. However, several of these holders hold nuclear material in amounts low enough to ask the European Commission to apply for exemption from safeguards under Article 37 in INFCIRC/193, or use nuclear material in such applications that it can be exempted under Article 36 in INFCIRC/193. If granted, they will not have to provide a site declaration or be subjected to inspections by the IAEA. By applying for derogations under Article 19 in the Commission Regulation (Euratom) no 302/2005 the reporting format and frequency can be simplified. If granted derogation the holder need only report when a change has occurred, instead of monthly, and they can do so using a special form instead of using ENMAS. They do however need to provide a report annually of their entire stock of nuclear material. Exemptions and derogations are not connected, i.e. being granted one of them does not mean being granted the other one. This means that there is in total four possible combinations here, where each combination comes with different rules and obligations (i.e. exemption + derogation, no exemption + derogation, exemption + no derogation, and no exemption + no derogation). All four combinations are represented in the group "Own MBA" in Sweden.

The "National LOF" in Sweden is a single Material Balance Area with the code WSWE. The members within this MBA are located all over Sweden. When a company or research institution acquires nuclear material (or as in most cases, discover that they already are in possession of nuclear material but had not reported it to the authorities) the first step SSM does is to include them in this MBA. Nuclear material transfers between different holders of the "National LOF" should not be reported to the European Commission; however SSM keeps track of the individual stock of nuclear material for all members of this MBA and records changes also within the MBA. When a change in inventory in or out of WSWE is reported to SSM it is subsequently reported to the European Commission using ENMAS. Exemptions and derogations can only be granted for an entire MBA, and since the inventory quite frequently changes for some of the members within WSWE no exemptions or derogations can be granted for any individual entity within this MBA. Therefore site descriptions are needed for all members. The first site declaration for this MBA was created and submitted in 2014 and the process leading up to this is described in section 5.

The "Catch-All MBA" is a common European Material Balance Area which gathers holders of small amounts of nuclear material in the Non-Nuclear Weapons States in the EU. There is a strict limit of how much material a member is allowed to have (specified in Annex I-G in the Commission regulation (Euratom) no 302/2005), and the total amount of nuclear material in the whole MBA must not exceed one effective kg (as stated in the Commission Recommendation of 15 December 2005, p. 35). The holders in this group are automatically granted derogation according to the Commission Regulation (Euratom) no 302/2005 and need only report to the European Commission when there are changes in the inventory. They are also exempted from IAEA safeguards and thus not subjected to inspections from the IAEA or under the obligation of providing a site declaration[4].

Table 1 shows a summary of some of the tasks and reports the holders in the different categories must do and submit.

4. Inspections of small holders

To ensure that requirements set up by the international organisations and SSM are met there is a need for communication, visits, and inspections. In the beginning of 2013 SSM started to put more resources into the work with holders of small amounts of nuclear material. The European Commission had previously announced that they would start to prioritize inspections of small holders and during 2013 they carried out a round of inspections of most holders in the category "Own MBA" in Sweden. The national work regarding small holders was further stimulated by a request from the IAEA to SSM to provide site descriptions according to the AP for all the entities within the National LOF WSWE.

	Own MBA	National LOF	Catch-All MBA
Keep inventory list	yes	yes	yes
Report changes in NM stock to SSM	yes	yes	yes
Report changes in NM stock to European Commission	yes	no (SSM reports)	yes
Report changes in NM stock using ENMAS	yes[1]	no (SSM reports)	no[2]
Report NM stock monthly	yes[1]	no (SSM reports)	no[2]
Site description (AP 2a(iii))	yes[3]	yes	no
Programme of activities	yes	no (SSM reports)	no[4]

Table 1: Summary of some obligations on holders of small amounts of nuclear material.

1 If derogation has been granted by the European Commission under Article 19 in the Commission Regulation (Euratom) no 302/2005, the frequency and format of the inventory reports can be modified from the standard way of reporting.

2 Members of the Catch all MBA are automatically granted derogation from the Commission Regulation (Euratom) no 302/2005.

3 If all NM in an MBA has been exempted from IAEA Safeguards following the provisions of Article 37 and 36 of INFCIRC/193 the site description is not required.

4 The requirements for Catch-All MBA members are further specified in the Facility Attachment "Safeguards agreement in connection with NPT, Subsidiary arrangements" [14] from 1985.

4.1 National inspections

An inspection carried out at a holder of small amounts of nuclear material should be prepared well in advance. Because of the vast differences between such companies and organizations that hold small amounts of nuclear material, the approach to ensure compliance with national and international regulations works best if it is tailor-made to fit the type of installation. It has been a learning curve for the national inspectors on how to best get the message across; on the one hand avoiding the use of too many abbreviations or technical jargon that unnecessary complicated things and on the other hand not simplifying too much or being too specific in instructing the installations.

Usually an inspection at a holder of small amounts of nuclear material that has not been visited in a long time (or has not been visited at all) starts with a phone call where the purpose of the inspection is explained. After that an email is sent summarizing the call and giving explicit instructions on what kind of preparations are expected from the holder before the visit. This can entail up-dating (or in some cases, creating) an inventory list (specifications are provided on what information such a list should contain), prepare shipping- or transport-documentation, and to make sure all nuclear material is available for id-checks and verification at the time of the visit. References to paragraphs in legal documents are also enclosed in the email.

During the inspection the inventory list prepared by the holder is compared with the inventory previously reported to SSM. Discrepancies are frequent. The search then begins to find out where material has been moved if it no longer can be found on the site, and update the registry at SSM with material that has either been found at the site or purchased without being reported. The responsibility for tracking the material is on the holder of the material, but in many cases SSM can be of help by having kept records of material transfers for decades. Often there is a need for a longer discussion and explanation of what should be reported and what type of information should be included. Then SSM verifies all material by number identification and sometimes by measurement by the use of an identiFINDER™ (HM5-type detector of gamma and neutrons). Depending on the category of the installation (see section 3.2) other information is requested to either be provided or updated, such as a basic technical characteristic (BTC) and/or a description of the site.

After the inspection there are usually a number of follow-up activities that need to be carried out and these are specified in a report written by SSM and distributed to the holder. The holders often need to further up-date their list of inventory items (LII) to comply with regulations, sometimes increase their efforts in locating documents supporting transfers of materials, or perform additional measurements on certain items to be able to declare them correctly. SSM often needs to update its registry of nuclear material as well as the registry on types of activities carried out by the holder and their contact information. The updated information is also reported to the European Commission.

In summary, time and effort spent during an inspection at an installation with very small amounts of nuclear material is not proportional if compared to a larger fuel cycle related facility, mostly due to the preparations and the follow-up activities. To ensure the best result follow-up inspections should be made, however only few such inspections have been carried out so far.

4.2 Inspections with international organizations

The Swedish Government has appointed SSM to accompany IAEA inspectors during international inspections in Sweden. In the case of inspections initiated by the European Commission and where the IAEA chooses not to participate, SSM makes a case by case decision to participate or not. In accordance with the intensified effort, inspections at small facilities are prioritized. These inspections follow the same format as for inspections in larger facilities. The books are audited and the internal book-keeping compared to the reported stock of nuclear material. The material is item identified and parts of it verified by non-destructive analysis (NDA) measurements. Because of the inadequacies encountered in many places SSM prepare the holders before the inspection as much as possible. If possible, a separate visit by SSM is scheduled and carried out before the international inspection. At the international inspections there is often less time to discuss matters of book-keeping, inventory lists, and reporting obligations. This is why visits made without international inspectors are a very important complement in ensuring compliance to regulations.

5. Site descriptions for the National LOF WSWE

The IAEA requested that Sweden (through the European Commission) would either ask for an exemption or provide a site description for the “National LOF” WSWE. It might seem like an easy request to meet, but it turned out to be quite challenging. The contacts with and inspections of the entities within this MBA had been few and far apart in time. Even finding out valid contact-information for some entities proved difficult.

It was quite quickly determined that an exemption from IAEA Safeguards was not a suitable option. Although the MBA holds an amount of nuclear material within the limit set out in Article 37 of INFCIRC/193 an exemption from safeguards would have been very impractical due to the frequent NM transfer in and out of the MBA. As a consequence SSM initiated a series of consultations with the European Commission and to some extent also with the

IAEA on how such a site declaration should be made and what information it should contain. There was also the question of how unified the declaration should be, because the nature of the activities performed by the entities within WSWE are quite diverse. It was decided that the level of details provided for the different members did not have to be unified, but instead depends on the activities within the specific installation. I.e. for metal scrap yards an overview map of the area is included and a brief explanation of the major activities performed on the site. For research institutions the site description needs to be more detailed with floor-plans, description of rooms and activities performed therein.

To simplify for the entities of WSWE a template of a site declaration was made by SSM for each installation where as much information as possible was already filled in. The holders only had to check and correct or in some instances provide some additional information. The ideal would have been to visit all installations before submitting the site declaration, to ensure its correctness and completeness, but unfortunately there was not enough time. It is instead an on-going task and the plan is to visit the installations in the near future.

After all the templates were checked and completed the site declaration was finally submitted to the European Commission in December 2014, and the first update to it was submitted in March 2015.

6. Experiences from working with small holders of nuclear material

In the past two years of working with holders of small amounts of nuclear material we have gained a lot of experience and learned a lot about different applications of nuclear material. However, some challenges have been encountered and they can be divided into two main parts. The first part can be summarised as communicating with and explaining to the small holders what their responsibilities as possessors of nuclear material are. The second part relates to the role of SSM and our mandate to interpret national and especially international regulations within the field of nuclear non-proliferation.

Communicating rules and regulations to the holders can be challenging. One of the surprises is that it is not always beneficial for the purpose of reporting if the holder has a great knowledge of nuclear physics. Every gram of nuclear material should be reported; when used in nuclear activities as well as for other purposes. A person with knowledge in the field might think that a couple of grams (or kilograms) of depleted or natural uranium cannot be used for anything illegal, therefor does not need to be reported.

Other types of holders, such as radiographers, are used to contacts with other organizational parts of SSM in applying for licenses to hold, use and transport radioactive isotopes. When the unit of Nuclear non-proliferation contacts them requesting information regarding the depleted uranium container in which the isotope is placed, it can lead to many misunderstandings. Another common misconception is that compounds containing natural uranium, depleted uranium, or thorium such as nitrates or acetates are automatically exempted from safeguards and safeguard reporting.

Even though the rules and regulations for all holders of nuclear material are the same, the prerequisites for fulfilling the obligations can be very different. At e.g. a nuclear power plant there are usually one or several people that have the dedicated task of keeping the inventory updated and managing the reporting duties of the plant. For a smaller installation responsibilities are often not formalized and the task of maintaining control of the nuclear material is not allowed enough time. When communicating with these holders this must be kept in mind. One must also understand that some words and abbreviations commonly used when communicating with larger facilities should be avoided or at least thoroughly explained.

The second part of challenges relates to the role of SSM in the international community of Nuclear non-proliferation. There is a need for a graded approach in applying regulations initially intended for facilities such as nuclear power plants on installations with small amounts of nuclear material. Often, a graded approach to requirements is not formalized, but employed nonetheless. SSM has the mandate to set terms or grant exceptions from national regulations, but the Commission Regulation (Euratom) no 302/2005 must be followed by all holders and the organization with a mandate to determine compliance is the European Commission. Without guidance documents on how to interpret rules and regulations from the viewpoint of a small holder it is difficult for SSM to help the installations on where to set the bar in trying to fulfil their obligations. Particular Safeguard Provisions (PSPs) or Facility Attachments (FAs) are not in place for the individual small holders in Sweden (with the exception of the ”Safeguards agreement in connection with NPT, Subsidiary arrangements” from 1985 which is valid for all members within the European “Catch All MBA”). When the holders ask questions such as “Is my answer to this question in the BTC specific enough?” or “Can we collect information about our experiments for a year and only report a re-batch at one time?” SSM can give advice based on previous experiences but cannot give definitive answers.

6.1 On-going work

In order to ensure correctness and completeness of the reports provided by the small holders themselves, or declarations that pass through SSM such as site declarations, inspections are needed. Since the start of 2013 SSM has increased its presence at these locations, and so far 13 of the 21 registered holders of small amounts of nuclear material have been visited at least once. Some have required several visits. Even when visits have not yet been possible to make, contact has intensified with all 21 of them in order to get updated information for e.g. BTCs and site declarations.

After a visit to a location and a meeting face-to-face where both we at SSM and the representatives from the installations have the opportunity to ask questions, the remainder of the communication runs much more smoothly. This is one reason why, even if the amounts of nuclear material is extremely small, a physical visit to such an installation is prioritized over some other tasks. The ambition is to perform inspections of as many of the small holders as possible and as soon as possible. However only a limited number of inspections are initiated at one time to allow proper follow-up since the post-inspection tasks are, in some cases, a lengthy process.

Prioritizing what installation to inspect is based on a number of factors; amount and type of nuclear material, application of nuclear material, perceived control by the installation of the nuclear material, its physical location and closeness to other installations.

An inspection can be triggered by the holder itself. With new staff at positions such as radiation safety experts SSM has sometimes been invited to talk about procedures for reporting and discussions on how to apply the rules and regulations to their specific activities. These visits are of course given high priority.

6.2 Plans for future work

To maintain correct and updated information on nuclear material inventory and information provided in BTCs and site descriptions we believe there is a need for regular follow-up activities. After the initial round of inspections has been completed we anticipate some sort of scheduled plan of activities, e.g. a rolling schedule of approximately 4-5 inspections annually and follow-up letters or phone calls biannually to all installations. Even though regulations state that changes should be reported without reminders we believe that to ensure complete and correct information some legwork is required from the side of SSM.

For most of the installations the nuclear material remains static for longer periods of time. However, a couple of installations use their material for experiments where the material form changes and material is relatively often sent and received, e.g. to and from collaborators. For these installations the plan is to focus inspections on their system for nuclear material accountancy and control as a means of ensuring that declarations of both nuclear inventory and technical capacity are complete and correct. This is especially important when there are several people involved and where there is a large turnover of personnel, e.g. at universities.

The next step is to search for, in a more structured way, installations which are in possession of nuclear material but are unaware of their reporting duties. This can be achieved by better co-operation and exchange of information within SSM, by various out-reach activities such as participating in meetings at relevant trade associations, and by using the knowledge gained over the past two years on what types of installations are most likely to possess nuclear material.

To make it easy to understand and to follow the rules and regulations applicable to holders of nuclear material there are plans to compose guide documents. These documents can be general nonetheless differentiated for the three categories we have in Sweden (described in section 3.2) or further tailored for specific types of installations.

As mentioned in section 3.1, installations which are in possession of nuclear material are also subjected to rules and regulations not related to nuclear non-proliferation and safeguards. To reduce the number of inspections at a specific installation and make better use of both their and SSM's time we are thinking about performing joint inspections, where e.g. nuclear non-proliferation, physical protection, and radiation protection is combined. In that way there will be expertise from several areas and SSM can provide the installations with a more collective view on what works well and where improvements can be made. The national inspectors can also learn from each other and after a couple of joint inspections cover an area which is usually not covered by that inspector.

7. Conclusions

Working to ensure nuclear non-proliferation with holders of small amounts of nuclear material, often not part of the nuclear fuel cycle is both challenging and time-consuming work, but also varying and fun. It demands a solid understanding of both national and international regulations, and the ability to transfer and translate them so that parties not familiar with non-proliferation can fulfil their obligations. When installations are found not to comply with rules and regulations, the reason seems almost always to be a lack of knowledge. Keeping contact by telephone calls, emails and making regular physical visits enables easier communication and ensures better compliance.

A formalized graded approach to rules and regulations, made in collaboration with the international organizations, would simplify working with small holders of nuclear material. This could be in the form of PSPs of FAs, or perhaps as a guide document similar to the "Guidance for States Implementing Comprehensive Safeguards Agreements and Additional Protocols" in the IAEA Service Series 21 [15].

8. Acknowledgements

The authors would like to acknowledge those at the unit Nuclear Non-proliferation and Transport at the Swedish Radiation Safety Authority who have reviewed and made valuable contributions to both this paper and to the work described herein.

The authors would also like to acknowledge the co-operation with the European Commission, with special thanks to Ali Zein, Timo Lindberg, Jean Schreiner and Christelle Charpentier, who have given good and speedy advice, help in interpreting regulations, and support in the EU reports provision.

9. Legal matters

9.1 Privacy regulations and protection of personal data

The authors 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.

9.2 Copyright

The authors agree 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 authors declare that their work/article is original and not a violation or infringement of any existing copyright.

10. References

[1] Treaty on the Non-proliferation of Nuclear Weapons (NPT); 1 July 1973.

[2] Dahlin G; *Sweden towards Integrated Safeguards*; *ESARDA Bulletin*, No. 43, December 2009.

[3] INFCIRC/234; *Agreement of 14 April 1975 Between the Government of the Kingdom of Sweden and the International Atomic Energy Agency for the Application of Safeguards in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons*; 18 February 1976.

[4] INFCIRC/153; *The Structure and Content of Agreements Between the Agency and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons*; 1 June 1972.

[5] INFCIRC/193; *The Text of the Agreement Between Belgium, Denmark, the Federal Republic of Germany, Ireland, Italy, Luxembourg, the Netherlands, the European Atomic Energy Community and the Agency in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons*; 14 September 1973.

[6] INFCIRC/193/Add. 8; *Protocol Additional to the Agreement between the Republic of Austria, the Kingdom of Belgium, the Kingdom of Denmark, the Republic of Finland, the Federal Republic of Germany, the Hellenic Republic, Ireland, the Italian Republic, the Grand Duchy of Luxembourg, the Kingdom of the Netherlands, the Portuguese Republic, the Kingdom of Spain, the Kingdom of Sweden, the European Atomic Energy Community and the International Atomic Energy Agency in implementation of Article III, (1) and (4) of the Treaty on the Non-Proliferation of Nuclear Weapons;* 12 January 2005.

[7] *COMMISSION REGULATION (Euratom) No 302/2005 of 8 February 2005 on the application of Euratom safeguards;* 8 February 2005.

[8] *Lag (1984:3) om kärnteknisk verksamhet*; 12 January 1984.

[9] *Förordning (1984:14) om kärnteknisk verksamhet*; 12 January 1984.

[10] *Lag (2000:140) om inspektioner enligt internationella avtal om förhindrande av spridning av kärnvapen*; 23 March 2000.

[11] *Förordning (2005:278) om inspektioner enligt internationella avtal om förhindrande av spridning av kärnvapen*; 4 May 2005.

[12] *SSMFS 2008:3 Strålsäkerhetsmyndighetens föreskrifter och allmänna råd om kontroll av kärnämne mm*; 30 January 2009.

[13] *Strålskyddslag (1988:220)*; 19 May 1988.

[14] *Safeguards agreement in connection with NPT, Subsidiary arrangements*; 30 January 1985.

[15] *Guidance for States Implementing Comprehensive Safeguards Agreements and Additional Protocols*; IAEA Service Series 21; 2014.

Methodological Aspects of the IAEA State Level Concept and Acquisition Path Analysis: A State's Nuclear Fuel Cycle, Related Capabilities, and the Quantification of Acquisition Paths

Lance K. Kim, Guido Renda, Giacomo G. M. Cojazzi

Institute for Transuranium Elements
European Commission Joint Research Centre,
Via Enrico Fermi 2749, Ispra 21027 (VA) Italy
E-mail: giacomo.cojazzi@jrc.ec.europa.eu

Abstract:

Within its State Level Concept (SLC), the International Atomic Energy Agency (IAEA) envisions a State Level Approach (SLA) for safeguards implementation that considers, inter alia, a State's nuclear and nuclear-related activities and capabilities as a whole when developing an annual implementation plan. Based on the assessed nuclear fuel cycle and related capabilities of a State, Acquisition Path Analysis (APA) identifies, characterizes, and prioritizes plausible routes for acquiring weapons-usable material to aid in safeguards implementation planning. A review of proposed APA methods and historical evidence indicates that assessments of pathway completion time can be fraught with uncertainty and subject to bias, potentially undermining safeguards effectiveness and efficiency. Based on considerations of theory and evidence, a number of methodological insights are identified to support consistent implementation and ongoing APA development. The use of algorithms to support APA and SLA processes in lieu of human judgement is a contentious issue requiring an evidence-based assessment and is also briefly discussed. This paper captures concepts derived primarily from open sources of information, including publications, presentations, and workshops on on-going APA development by the IAEA and various Member States Support Programs (MSSP) as well as relevant work found in the open literature. While implementation of the SLA has begun for a number of States, these SLAs are being updated and developed for other States. In light of these ongoing developments, the topics covered here should be considered a snapshot in time that does not reflect finished products and does not necessarily reflect official views.

Keywords: Safeguards, State Level Concept, Acquisition Pathways Analysis

1. Introduction: The IAEA State-Level Concept for Improving the Effectiveness and Efficiency of International Safeguards Implementation

International Atomic Energy Agency (IAEA) safeguards continue to evolve to respond to new challenges. With the introduction of the Additional Protocol (AP), approaches for detecting the diversion of material and the misuse of declared facilities have been complemented by additional measures to strengthen the detection of possible undeclared activities. The IAEA's State Level Concept (SLC) envisions a holistic approach to nuclear safeguards considering the State as a whole to tailor safeguards implementation to the State. Within the SLC, the IAEA envisions a State Level Approach (SLA) for safeguards implementation that considers a State's nuclear and nuclear-related activities and capabilities to meet generic safeguards objectives – the detection of diversion, misuse, and undeclared material or activities.[1], [2]

Within the State Level Approach (SLA) for safeguards diagrammed in Figure 1, Acquisition Pathways Analysis (APA) estimates the time necessary to complete plausible routes to weapons-usable material based on all information available on a State. Safeguards measures are selected to satisfy technical objectives with a frequency and intensity dependent upon path completion times and the effectiveness and efficiency of safeguards measures. A SLA for safeguards implementation is then developed and executed through an annual implementation plan.[1]–[3]

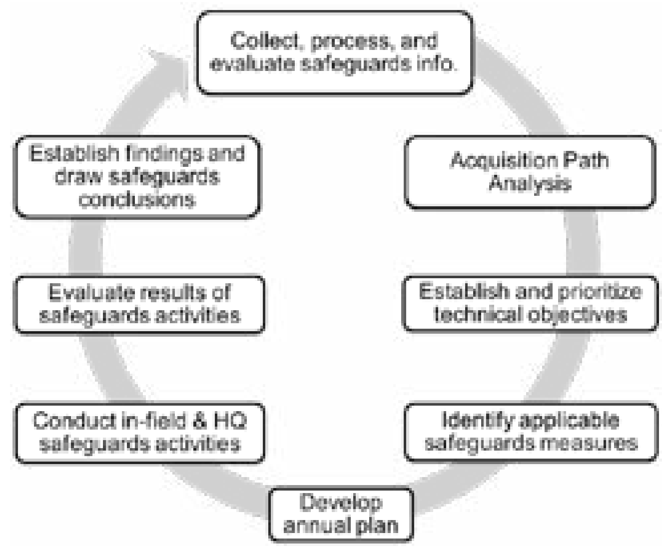


Figure 1: Flow chart of processes supporting State-level safeguards implementation, adapted from [1]

Improving the effectiveness and efficiency of safeguards implementation in a non-discriminatory manner are among the key principles of safeguards implementation under the SLC.[1] However, attaining these aspirations is not straightforward. While the IAEA envisions lower in-field verification efforts in States where a Broader Conclusion (BC) can be drawn that all nuclear material is safeguarded and remains in peaceful uses[1], [4], assuring the absence of undeclared activities may be costly to achieve as, "There are no general answers to the question of how much verification is enough…".[5] Perceived over-spending in States with substantial nuclear infrastructure "generally accepted as presenting low proliferation risk" [6] may reflect non-technical (e.g., security environment, form of government, etc.) rather than technical State Specific Factors (SSF), potentially engendering accusations of discrimination and demanding a level of transparency that some consider lacking.[7] While deliberately provocative, these scenarios underscore the importance of an evidence-based approach to ensure the non-discriminatory application of effective and efficient safeguards.

To begin to address some of the issues affecting safeguards effectiveness and efficiency, and non-discrimination goals, the role of State Specific Factors (SSF) is particularly important to understand as they form the basis for tailoring safeguards implementation to a particular State. The IAEA has identified six categories of SSFs (Table 1) that influence the design, planning, conduct, and evaluations of safeguards activities.[1] Though the IAEA has stated that this list of factors is "exhaustive", [2] the absence of some factors (e.g., form of government, multinational control, etc.[8], [9]) potentially misses effectiveness and efficiency gains. Expanding upon the IAEA's diagram, the following diagram of SLA program theory [10] in Figure 2 reflects the premise that differentiating States on the basis of SSFs leads to improvements in safeguards effectiveness and efficiency. Though by no means complete, this diagram broadly outlines the influence of SSFs on SLA processes.

While many of the IAEA's SSFs appear related to relatively uncontroversial technical efficiencies (e.g., new safeguards measures) or implementation issues (e.g., cloudy spent fuel pools), opinions differ on the objectivity of these factors (Table 1). One commenter characterized these SSFs as based on "indisputable facts", or "more open to disagreement", but still based on shared implementation experience without "inherently subjective" factors such as the State's political situation, intentions, or regional stability.[11] However, another commenter characterized four of the six SSFs as "hardly quantifiable" or "discretionary".[9] On the other hand, the nuclear fuel cycle and related capabilities of the State are said to be "well quantifiable" [9] and based

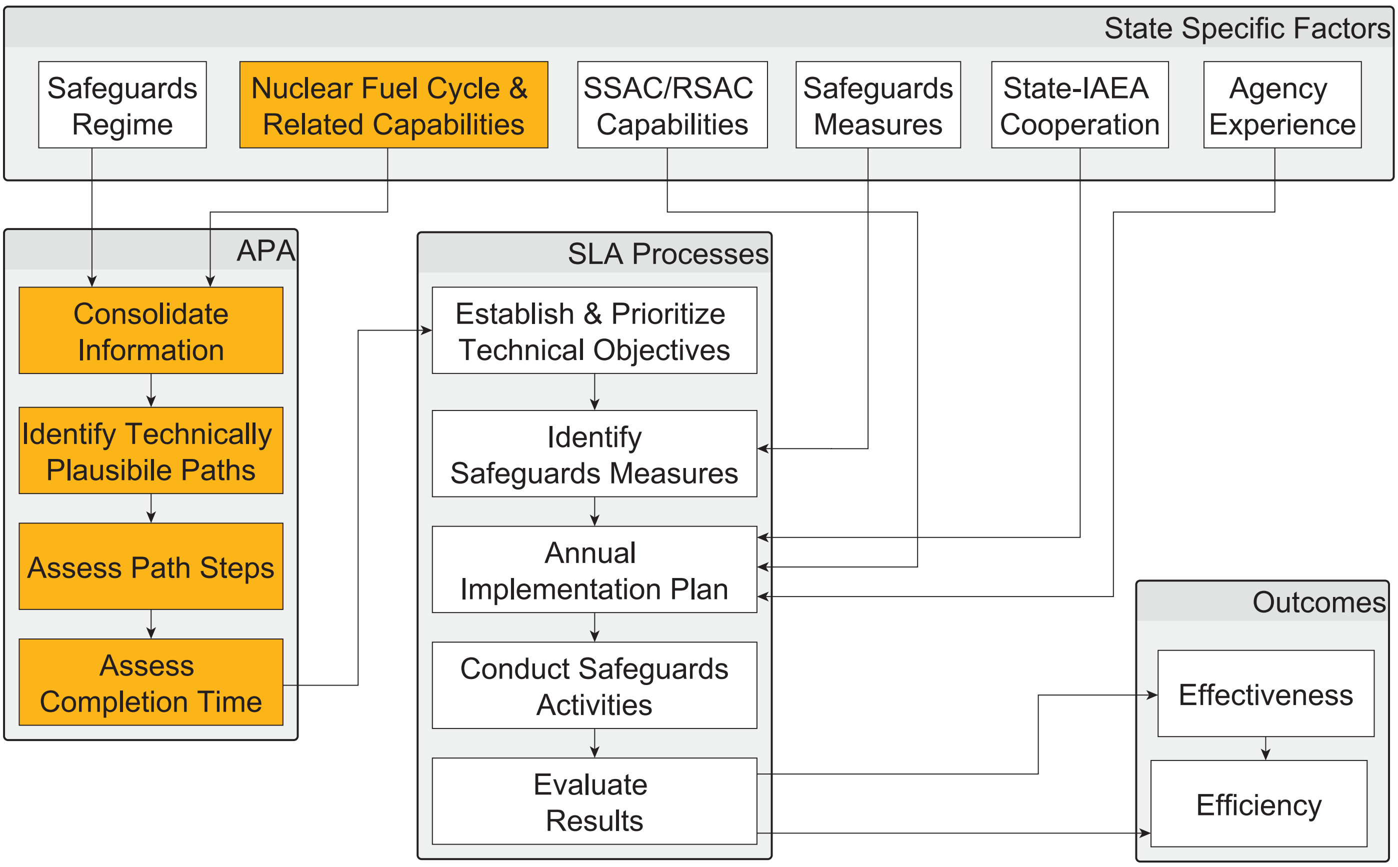


Figure 2: Notional sketch of SLC program theory expanding upon the IAEA's flow chart (Figure 1) to link State Specific Factors (Table 1) to Acquisition Pathways Analysis and effectiveness and efficiency measures. As highlighted, this paper focuses on the second State Specific Factor.

State Specific Factors [1]	Medici, 2014 [9]	Burton, 2014 [11]
"The type of safeguards agreement in force for the State and the nature of the safeguards conclusion drawn by the Agency"	"well quantifiable"	"…indisputable facts and thus not open to misinterpretation or ambiguity."
"The nuclear fuel cycle and related capabilities of the State"	"well quantifiable"	"…indisputable facts and thus not open to misinterpretation or ambiguity."
"The technical capabilities of the State or regional system of accounting for and control of nuclear material (SSAC/RSAC)"	"hardly quantifiable"	"…more open to disagreement between the IAEA and the State, but are still based on the shared experience in implementing safeguards in the State."
"The ability of the Agency to implement certain safeguards measures in the State (e.g. remote monitoring, unannounced/ short notice inspections)"	"discretionary"	"…indisputable facts and thus not open to misinterpretation or ambiguity."
"The nature and scope of the cooperation between the State and the Agency in the implementation of safeguards"	"discretionary"	"…more open to disagreement between the IAEA and the State, but are still based on the shared experience in implementing safeguards in the State."
"The Agency's experience in implementing safeguards in the State"	"discretionary"	"…more open to disagreement between the IAEA and the State, but are still based on the shared experience in implementing safeguards in the State."

Table 1: Summary of the quantifiability and objectivity of State-Specific Factors, derived from [1], [9], [11]

on "…indisputable facts and thus not open to misinterpretation or ambiguity."[11]

Despite the apparent consensus on the quantifiability of a State's nuclear fuel cycle and related capabilities, this SSF is particularly important to understand as it potentially reduces prior conservatisms. Whereas timely detection goals had been based on the assumed presence of undeclared capabilities, new approaches may be based on the presumed absence of undeclared capabilities or on the estimated completion time of hypothetical pathways. As untimely detection could result from an underestimation of a State's capabilities, the uncertainties injected into the design of safeguards need to be understood and characterized.

Adopting a critical view, a number of methodological insights to the APA process are identified from consideration of theory and evidence to support ongoing APA development and implementation. This study is based on an examination of several Acquisition Pathways Analysis (APA) processes proposed by the IAEA and its Member States as well as related literature on the management of complex engineering projects and the estimation of nuclear latency. In addition to this theoretical exercise, historical nuclear intelligence assessments and nuclear project timelines are reviewed in search of evidence. It also builds upon previous contributions by the European Commission Joint Research Centre in the area of methodological considerations for the APA process and the potential of open source information for supporting the APA.[12]–[14]

2. A State's Nuclear Fuel Cycle and Related Capabilities: How Quantifiable is Completion Time?

To address the degree to which a State's nuclear fuel cycle and related capabilities can be objectively quantified, this section explores how information about a State's nuclear fuel cycle and related capabilities is used by Acquisition Pathways Analysis (APA) to prioritize paths based on completion time. As the technical backbone of the SLA process, the APA estimates the ease and speed by which a State might acquire weapons-usable material through a four stage process described below.

- Information Collection: "Consolidating information about the State's past, present, and planned nuclear fuel cycle-related capabilities and infrastructure"
- Path Identification: "Identifying and visually presenting technically plausible acquisition paths for the State"
- Path Characterization: "Assessing acquisition path steps (State's technical capabilities and possible actions) along the identified acquisition paths"
- Path Prioritization: "Assessing the time needed to complete a technically plausible acquisition path" [3]

These four stages are discussed in series in the following sections, identifying informational and analytical uncertainties that affect estimates of completion time.

2.1 Information Collection

Information asymmetries between the IAEA and the State are particularly important to characterize as imperfect information can lead to the misallocation of safeguards resources. While more information is available about a State under the Additional Protocol (CSA+AP) than one with solely a Comprehensive Safeguards Agreement (CSA-only), it may be more important to contemplate the unknowable rather than emphasize what is known. As discussed in an earlier paper on the use of open source information in the APA[13], while information is readily available to the IAEA about a State's declared nuclear capabilities, information may be considerably more opaque when considering potential misuse and undeclared paths arising from low signal-to-noise ratios and deliberate actions to conceal information or spread disinformation.[15], [16] As the pendulum of advantage swings from competition between hiders and seekers, information asymmetries may be difficult to characterize at any particular moment. Analysts must confront the possibility that the evidence in hand may be incomplete, unreliable, ambiguous, and even deceptive. While the searchlight may be broader and brighter, the item sought may still be shrouded in darkness.

2.2 Path Identification: Technical Plausibility as a Screening Criterion

Overall, the process of path identification is based on solid foundations, leveraging the IAEA's Physical Model [17] to identify safeguards-relevant flows of materials. While there are some issues to resolve for effective path identification (e.g., the choice of material- or facility-centric views, additive or subtractive path identification), the definition of a technically plausible pathway is perhaps the most significant conceptual issue affecting path identification. As defined by the IAEA, a plausible path is one that, "…a State could, from a technical point of view, acquire at least one significant quantity of weapons-usable nuclear material within five years…".[3] Defined in this manner, technical plausibility is essentially a screening criterion to "prune" pathways that are deemed too far off in the future to warrant further consideration in the APA, presumably to reduce analytical burdens. As technical plausibility is essentially a preliminary estimate of completion time, the analytical process is further discussed in detail below when considering path prioritization.

At this stage of the APA process, the use of technical plausibility as a pruning criterion warrants further consideration. Should this determination be incorrect, inadequate attention to paths that are indeed plausible may lead to untimely detection. As the demarcation between plausible and implausible appears somewhat arbitrary, a sounder technical basis is desirable to avoid excessive pruning. For example, the five-year criterion can be construed as a detection goal for *semper vigilans* safeguards measures (such as those associated with the State evaluation processes including the monitoring of open source information, third-party sources of information, etc.). For a given measure with an annual detection probability, P_{annual}, the probability of detection after t years is, $P(t) = 1\text{-}(1\text{-}P_{annual})^t$ assuming independence of detection events. A safeguards measure then requires a 37% annual probability of detection to achieve a 90% probability of detection within five years or an annual detection probability of 21% over 10 years. These goals, however, may be largely aspirational as the probability of detecting undeclared activities may elude quantification.

2.3 Path Characterization

Following the identification of plausible paths, path characterization is largely a structured collection of information about a State's technical capabilities to complete a defined path. Relevant details of a State's capabilities depend on the type of acquisition path step. The IAEA identified five types of path steps: indigenous production, diversion, misuse, clandestine production, and undeclared import. In the case of diversion, relevant technical details include inventories of material and their characteristics. In the case of potential undeclared activities, information includes assessments of the State's knowledge, R&D, current capabilities to manufacture or purchase equipment, and experience with operating related processes. State actions (i.e. proliferation scenarios) are then identified and assessed for each plausible acquisition path step. In the case of diversion, State actions include the diversion of spent fuel assemblies with replacement by dummies and the diversion of spent fuel rods through the disassembly of fuel assemblies. Misuse includes such actions as using excess production capacity and making concealed modifications or upgrades to a facility. Clandestine paths are assessed with respect to the actions necessary to acquire the missing capability through indigenous development and/or importation.[3]

2.4 Path Prioritization: Complexities of Estimating Completion Time

The final stage of the APA is perhaps the most analytically challenging element of the APA process. Based on preceding stages, the "…ease (technical capabilities) and speed by which a State could acquire one significant quantity of nuclear material using that path" are assessed to estimate path completion time.[3] For the purposes of safeguards planning, path completion time is essential for establishing timely detection goals. However, completion time should not be mistaken as a measure of path attractiveness or likelihood as "…the most likely path is not necessarily the quickest path".[18]

Approaches such as the Program Evaluation and Review Technique (PERT) [19] have been used to plan complex projects, including the Hanford Engineer Works during the Manhattan Project [20], and to estimate the time necessary for a nascent nuclear weapons program.[21] Complexities arise as analysts must consider several factors including, "…resources, in the form of dollars, or what 'dollars' represent – manpower, materials, and methods of production, technical performance of systems, subsystems, and components, and time."[19] In a simplified view of these complexities, completion time can be thought of as a combination of the ease of the task relative to the capabilities and resources of a State to accomplish the task.

2.4.1 Assessing Ease: Intrinsic Technical Difficulty and State Capabilities

While the IAEA appears to equate "ease" with the technical capabilities of the State, the intrinsic ease of the task itself also requires characterization, without which, completion time cannot be estimated. As defined in the Generation IV International Forum's Proliferation Resistance and Physical Protection (PR&PP) evaluation methodology, the intrinsic technical difficulty of a path step (defined as "technical difficulty" by PR&PP[1]) is, "The inherent difficulty arising from the need for technical sophistication, including material-handling capabilities, required to overcome the multiple barriers to proliferation."[22] Analytical models demonstrate that path rankings are sensitive to the ratio between the intrinsic technical difficulty and a State's capabilities, underscoring the importance of "realistic assumptions" regarding both factors.[23]

Characterization of Intrinsic Technical Difficulty: While intrinsic technical difficulty is captured to some extent when identifying and characterising paths, "realistic assumptions" about intrinsic technical difficulty should be stated explicitly to avoid ambiguity. For example, as reported in a recent review study, some claim that "…all enrichment techniques demand sophisticated technology in large and expensive facilities", suggesting that enrichment is out of reach of all but the most capable States. Others have suggested that a small centrifuge plant is "…feasible for countries with no prior experience, 'that possess relatively little technical skills and which have relatively little industrial activity'".[24] As illustrated by several studies [18], [21], [22], [25]–[29], quantitative descriptions may help narrow differences between competing views by explicitly stating proliferation costs, labor requirements, necessary materials, and physical processes necessary to complete a task.

"Related Capabilities" may be Very Broad: The delineation between a State's technical capabilities related and unrelated to the nuclear fuel cycle requires definition. It may be the case that "related capabilities" important to understanding a State's ability to pursue pathways may be quite broad – so broad that even non-nuclear States may have plausible pathways. After all, nearly three-quarters of a century ago, only seven years elapsed between the discovery of fission and the first use of a nuclear weapon. While these may have been extraordinary early efforts, "What was once exotic is now pedestrian."[24] These linkages are further evidenced by the correlation between proliferation decisions and, *inter alia*, general economic development measured by factors such as gross domestic product and the production of energy and steel.[30], [31] As just one example, States that exploded or deployed nuclear weapons had an average GDP per capita of $8000 (in 2015 U.S. dollars) and many States were below that average.[30]

The use of economic status may be contrary to IAEA guidance against the use of SSFs to "rate or grade States" and may be considered a "political or other extraneous consideration".[1] However many SSFs could also be misused to rank States and economic status may be rationally related to safeguards effectiveness and efficiency improvements. For instance, economically developed States may prefer indigenous pathways to better maintain secrecy and are thus less likely to be detected by monitoring imports.[32]

2.4.2 Assessing Speed: Historical Evidence and Sources of Analytical Uncertainty

After defining intrinsic technical difficulty in sufficient detail, engineering management methods can estimate completion time in light of a State's capabilities and resources. Estimating how fast or how slow contends with sources of analytical uncertainty that can amplify informational uncertainties. The size and direction of these uncertainties depend on the balance that is struck between false positive and false negative errors.

History of Misestimation May Undermine Plausibility Determinations: For undeclared paths, a retrospective study of U.S. intelligence estimates concluded that the nuclear capabilities of States have tended to be overestimated i.e. States have tended to acquire capabilities later than expected.[33] This tendency for early warning, while potentially alarmist, is less susceptible to false negative surprises.

However, by excluding estimates of weaponization phases, our examination of the cited cases suggests that foreign nuclear fuel cycle capabilities tend to be underestimated i.e. States have acquired nuclear fuel cycle capabilities sooner than expected. Of the 35 cases that did not involve weapons development and testing, 13 were underestimated, 13 were correct, and nine were overestimated. Quantitative time estimates are available for nine of these 35 cases with an average underestimate of approximately five years (Table 2). A number of misestimates, such as the surprise revelation of the Argentinian enrichment capability, are not reflected in this tally as timing information is not readily quantified.[33]–[35]

[1] The PR&PP methodology evaluates the proliferation resistance of a nuclear energy system in the basis of the response of the system to challenges. The response is assessed through pathways analysis. Diversion/misuse pathways are characterized by six measures: technical difficulty, cost, time, material type, detection probability, detection resource efficiency.[22]

Direction	Cases	Quantified Cases	Average Error (Years)
Underestimated	13	5	4.8
Correct	13	1	-
Overestimated	9	3	1.8

Table 2: Estimates of foreign nuclear fuel cycle capabilities from a recent study of US intelligence estimates, derived from [33]

This history of misestimation by intelligence agencies suggests that even a well-structured process may lead to significant underestimates of a State's nuclear fuel cycle capabilities. While it is difficult to generalize from this small sample size and perhaps even more difficult to extrapolate to modern verification and evasion techniques, the potential for underestimation may be large enough that capabilities judged to be implausible within five years may already exist within a State. While the apparent discrepancy between estimates of fuel cycle and weaponization stages requires further study, possibly with more rigorous techniques [36], biases and analytical uncertainties are discussed below as potential contributors to these misestimates.

Biases and Misestimation: The same review of the history of nuclear intelligence identified political, cultural, bureaucratic, and organizational distortions that contributed to misestimation (Table 3). Cultural biases may be particularly concerning as they may engender accusations of discrimination and also lead to substantive errors from e.g., "… failure[s] to understand an economic system".[33] While the potential size of cultural biases is unstated, they appear to act in both directions, contributing toward both underestimation and overestimation (Table 4). While international organizations may be less susceptible to some systemic cultural biases than individual States, they remain vulnerable to other biases such as policy and budgetary pressures that may lead analysts to downplay challenging scenarios.

Analytical Uncertainties Amplify Informational Asymmetries: From considerations of theory, analytical processes may have also contributed to misestimates. In the context of the APA, two principal sources of uncertainty are important to recognize: informational and analytical.[13] As discussed earlier, analysts must first consider the reliability of the information in hand and understand the potential for unobtainable information. Even assuming certain information, the certainty of path completion times estimates vary by the type of pathway and the degree to which factors of production are fixed or variable. In the parlance of production economics, factors of production (e.g., land, labor, capital equipment) are fixed if they are not readily altered over the short-run while all factors of production are variable in the long-run.[37] Along this continuum, factors of production are largely fixed in diversion scenarios that exploit declared infrastructure and well understood means of moving nuclear material. Analytical uncertainties are greater for misuse paths as factors of production are more variable than diversion and a State's capabilities are more salient. Clandestine paths have considerably more degrees of freedom with the potential to produce widely diverging estimates depending on the set of assumptions. And so while a State's declared capabilities may be "well quantifiable", undeclared capabilities are less certain.

Completion Time Depends on Path End State and Task Scheduling: Some of this potential divergence is addressable with guidance on the degree of conservatism to be exercised when specifying end states and path step schedules. As mentioned earlier, estimates of time depend upon the intrinsic technical difficulty of the path end state. For example, for a State known to have conducted basic

Type	Distortion Hypothesis
Political	• The ideology of the executive may encourage or promote those estimates that conform to the desired view • Policy initiatives, past, present, and future, can affect estimates (e.g. existing policy makes difficult or precludes objective analysis, whether logistically or psychologically, not enough importance attached to an area of geography or analysis, likelihood of major action resulting from estimate) • Likelihood of disclosure / politicization of estimate
Cultural	• Cultural biases create mistaken assumptions of capabilities • Misestimating intent / motives / resolve of subject State • Analysts misinterpret the involvement of outside sources
Bureaucratic	• Multiple advocacy among agencies causes compromise and/or domination • A fragmented bureaucracy stalls the dissemination and aggregation of useful data
Organizational	• Data overwhelms the analytic system, signals not separated from noise • Preference for secret over open sources [and vice versa?] • Recent experience with intelligence failures • Mistaken induction / conceptual rigidity: assumptions derived from historical experiences may not apply

Table 3: Categories of intelligence distortion hypotheses, summarized from [33]

State	Program Period	Direction of Cultural Bias	Description
Germany	1941-1945	Overestimation	"Culturally, beliefs about the abilities of the German scientists and a motivated misinterpretation of the slightly delayed publications where absence of evidence was considered evidence also contributed to distortion."
France	1954-	Overestimated	"… it is likely that cultural biases were positive in the French case, causing analysts to downplay the probability of inevitable problems and pushing estimates forward."
Israel	1955-	Underestimated	"… underestimated Israeli technical capabilities, as it was believed that Dimona could not be completed without US or French assistance"
China	1956-	Underestimated	"… underestimation of native Chinese production capabilities was a major factor in skewing earlier estimates, which expected reliance on Soviet assistance."
Iraq	1973-1991	Underestimated	"The program's underestimation was driven by a prior underestimation of Iraqi manufacturing capabilities, as evidenced by the expected reliance on foreign sources."
Libya	1970-2003	Underestimated	"Prior views of Libya's incompetence (although justified) may have contributed to the six-year gap between Libya's decision to seek a nuclear program through assistance from the A.Q. Khan network and the CIA reports of Libyan attempts to acquire materials from abroad."

Table 4: Examples of cultural biases impacting assessments of a State's technological capability, table derived from [33]

radiochemistry experiments, a large production reprocessing facility that takes a decade to complete may be implausible, but a small pilot reprocessing facility is nearly plausible, and a "quick and dirty" reprocessing system may be faster yet (Table 5).[27], [28], [38] Evaluating a range of technological options of varying intrinsic technical difficulty challenges analysts to consider scenarios that might otherwise go unnoticed or downplayed.

Scheduling can also dramatically impact completion time. As evidenced by a Gannt chart timeline of the Hanford facility from the Manhattan Project (Figure 3), a project that might have spanned nearly two decades in peacetime was claimed to have been substantially accelerated to about a third of that time during wartime by performing steps more quickly and in parallel [20] – though this claim cannot be substantiated due to the lack of a counterfactual alternative history. Another study estimated that doubling available resources could expedite the completion of an aerodynamic enrichment project from 338 weeks down to 260 weeks.[21] From these studies, substantial compression of path completion time appears plausible.

Omitted State Specific Factors: Schedule compression arising from the exigencies of war illustrates the potential impact of non-technical State specific factors (e.g., budgetary resources, proliferator goals, organizational issues [21], [39]) that have been formally omitted from consideration. Though engineering management methods can be used to estimate pathway time, such estimates are often wrong without accounting for motivational factors and institutional barriers that may hasten or slow progress. As noted by a study on latency, "…if one uses [an engineering management] approach…, the time predicted for a State to develop its first nuclear device tends to be incorrect" as "…pathway decisions are determined by various

Technology	Estimated Time to "Quick and Dirty" Facility [27], [28]	Average Time to Pilot Plant (years) [38]	Average Time to Production (years) [38]
Enrichment (diffusion)		-	6
Enrichment (centrifuge)		8	14
Enrichment (EMIS)		2	3
Enrichment (chemical)		6	11
Enrichment (aerodynamic)		7	18
Enrichment (laser)		-	-
Graphite-moderated production reactors		1	2-11
Heavy-water-moderated production reactors		1	2-6
Research reactors			4-5
Reprocessing	4-6 months	6	10

Table 5: Estimated and historical timelines for various fuel cycle technologies, derived from [27], [28], [38]

motivations and institutional impediments that often outweigh the pure engineering resource management decisions."[25]

These omitted factors may be perceived as subjective and discriminatory if used to differentiate States. Nonetheless, guidance on these omitted State specific factors may be necessary to limit analyst discretion and assure uniform implementation across States. For example, analysts may be instructed to make assumptions that minimize path time by assuming that all states are motivated, have large budgetary resources, and competently manage the pursuit of technologies that are easier than commonly perceived. It remains to be seen to how the frequency and intensity of safeguards implementation under the SLC will differ from current practices that assume the presence of undeclared activities when establishing quantity and timeliness goals.

3. Summary of Methodological Insights

The potential pitfalls of Acquisition Path Analysis reflect the complexity of estimating path completion time – a process that may lead to inconsistent assessments and may leave safeguards vulnerable to unpleasant surprises. To summarize the preceding sections, a number of process needs are identified in recognition of these potential pitfalls to support more consistent implementation of the APA including:

- Understanding the extent of information asymmetries
- Developing a basis for the technical plausibility criterion
- Characterizing intrinsic technical difficulty
- Identifying the related technical capabilities of a State
- Recognizing the potential for bias
- Characterizing analytical uncertainties
- Issuing guidance on omitted State specific factors

3.1 Model Pathway Approaches May Improve Consistency and Transparency

Incorporating some of these insights into model or generic pathway safeguards approaches may support more consistent implementation of the APA and the SLA. Analogous to model facility approaches that are modified on the basis of facility-specific information, modifications to a model pathway approach could arise from the APA through the channels of quantity and timeliness goals in addition to other SSFs (e.g., the capabilities of the SSAC and safeguards implementation issues). Model pathway approaches may also clarify the impact of other SSFs on safeguards implementation, for example, by defining modifications to path priorities for States with the Broader Conclusion.[40]

3.1.1 Algorithms or Expert Judgement?

Model pathway approaches may be considered overly prescriptive as they are potentially translated into an algorithmic approach leaving little room for expert judgement. Any algorithmic result, such as those based upon game theoretic models [41], should be treated with caution as they are normative and may not accurately describe a State's behavior. However, human analysts should receive equal if not greater scrutiny.

Algorithms may outperform humans in both "low validity" and "high validity" environments. In "low validity" environments, humans have difficulties detecting weak causal linkages, make inconsistent decisions, and yet may develop an illusion of skill. Predicting the future value of stocks and long-term political developments are examples of "low-validity" activities. In "high-validity" environments with frequent objective feedback, humans can develop true skill, but algorithms might still outperform humans who are subject to lapses in attention. Medicine and firefighting are examples of situations where true skill can develop.[42]

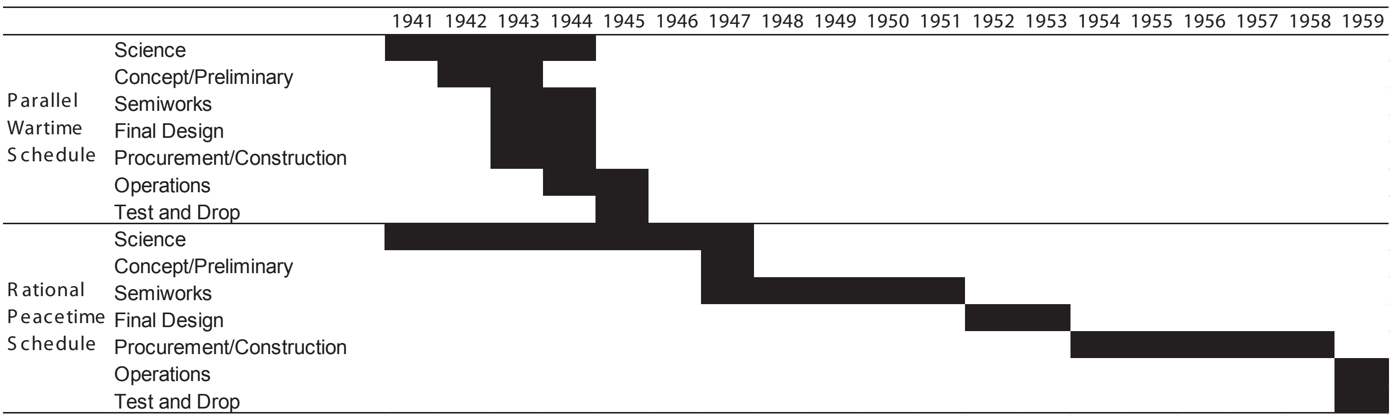


Figure 3: Gantt chart timeline of the Hanford facility from the Manhattan Project in wartime and a hypothetical peacetime scenario, derived from [20]

Where the APA lies on this spectrum of validity is debateable. Though safeguards analysts can learn established practices, the relationship between cause and effect may be vague as the effectiveness of a safeguards plan on the behaviour of States is difficult to evaluate objectively. Lacking the “gold standard” of program evaluation – a randomized controlled study [10] – evaluations of safeguards effectiveness are inherently subjective. A report written in 1995 by the U.S. Office of Technology Assessment notes that, “Despite the objective, systematic way in which the IAEA nuclear safeguards system has been codified and implemented…the underlying judgment as to what the safeguards system needs to be able to do and how well it needs to do it is inherently a subjective one.”[43] Coupled to the rarity of proliferation events, the short tenure of safeguards analysts, and the limitations of institutional memory, a weak causal feedback loop suggests that safeguards implementation planning may resemble a “low validity” type of activity.

Algorithms will face similar issues without adequate data for verification and validation, but may eventually outperform human analysts on average. However, while the consistency of algorithms may contribute to differentiation without discrimination, even algorithms may exhibit discriminatory behavior as they are designed by humans or trained by historical data.[44] – though discrimination-aware algorithms are under development.[45] Objective evaluation of competing approaches is essential to overcome algorithm aversion – the preference for human analysts even when algorithms are demonstrably less prone to error.[46]

4. Summary

While the APA process appears conceptually sound at a high level, considerations of theory and evidence suggest that significant uncertainties may be encountered when estimating path completion time based on State’s assessed nuclear fuel cycle and related capabilities. An understanding of these uncertainties is essential for striking a balance between false negative and false positive errors when developing a safeguards implementation plan. Methodological insights identified to support continued APA development reflect the complexities of estimating path completion time that may lead to inconsistent assessments and may leave safeguards vulnerable to unpleasant surprises. Potential process improvements essentially boil down to the need for a “surprise-sensitive” [33] safeguards planning approach, including an APA process that postulates a highly motivated, resourceful State that rapidly pursues pathways that are easier than assumed and where information asymmetries favor the “hider”. These types of analytical assumptions might be incorporated into model pathway approaches to promote consistent implementation, constrain analyst discretion to limit potentially discriminatory biases, and clarify how safeguards implementation might change under the SLC.

The use of algorithmic approaches supporting the consistent implementation of safeguards will continue to be an area of debate. While algorithmic consistency may contribute to non-discrimination goals, decision models are an imperfect science. On the other hand, expert judgement is also imperfect. A resolution to the debate ultimately requires an evidence-based, not faith-based, approach evaluating the impacts of safeguards on a State’s behaviour and examining discriminatory impacts. Such evidence is likely not forthcoming as a randomized controlled study of safeguards effectiveness is impractical. Nevertheless, some insights may be derived by considering the validity of the safeguards environment. Such thinking, as noted in early work on collaborative human-machine approaches for safeguards, [47] may better harness the consistency of algorithms while bringing human reasoning to bear where it is needed most.

This paper captures concepts derived primarily from open sources of information, including publications, presentations, and workshops on on-going APA development by the IAEA and various Member States Support Programs (MSSP) as well as relevant work found in the open literature. While implementation of the SLA has begun for a number of States, these SLAs are being updated and SLAs will be subject to “progressive development…for other States in the future.”[48] In light of these ongoing developments, the topics covered here should be considered a snapshot in time that does not reflect finished products and does not necessarily reflect official views.

5. Acknowledgements and Disclaimer

The work here presented is being carried out within the project “Innovative Concepts and Methodologies for Nuclear Safeguards” funded within the European Commission (EC) Euratom Horizon 2020 Research and Training Programme, also as a contribution to the EC Support to the IAEA task JNT C 1871: “Acquisition Path Analysis Methodology and Software Package”. The views expressed in this document are those of the authors and do not necessarily represent the official views of the European Commission.

6. References

[1] "The Conceptualization and Development of Safeguards Implementation at the State Level," International Atomic Energy Agency, Available at: http://www.isisnucleariran.org/assets/pdf/GOV201338.pdf, GOV/2013/38, Aug. 2013.

[2] "Supplementary Document to the Report on the Conceptualization and Development of Safeguards Implementation at the State Level (GOV/2013/38)," International Atomic Energy Agency, Available at: https://armscontrollaw.files.wordpress.com/2014/09/iaea-state-level-safeguards-document-august-2014.pdf, GOV/2014/41, Aug. 2014.

[3] T. Renis, Y. Yudin, and M. Hori, "Conducting Acquisition Path Analysis for Developing a State-Level Safeguards Approach," presented at the INMM, 2014.

[4] International Atomic Energy Agency, "IAEA Safeguards Glossary," Austria, IAEA/NVS/3, 2002.

[5] A. S. Krass, *Verification: How Much is Enough?* Stockholm International Peace Research Institute, 1985.

[6] J. Carlson, R. Leslie, P. Riggs, and A. Berriman, "Back to Basics - Re-Thinking Safeguards Principles," presented at the Annual Meeting of the Institute of Nuclear Materials Management, Phoenix, Arizona, 2003.

[7] T. McIntosh and W. Burr, "International Atomic Energy Agency Lacks Transparency, Observers and Researchers Say," *The National Security Archive*. [Online]. Available: http://nsarchive.gwu.edu/nukevault/ebb512/. [Accessed: 28-Apr-2015].

[8] M. Hibbs, "The Plan for IAEA Safeguards," *Carnegie Endowment for International Peace*, 20-Nov-2012. [Online]. Available: http://m.ceip.org/publications/?fa=50075. [Accessed: 06-Mar-2014].

[9] F. Medici, "Traditional/Integrated Safeguards vs. State Level Concept: Where is the IAEA heading to?," presented at the ESARDA Joint Meeting on IAEA State Level Concept, Ispra, Italy, 12-Nov-2013.

[10] P. H. Rossi, M. W. Lipsey, and H. E. Freeman, *Evaluation: A Systematic Approach*, 7th ed. Sage Publications, Inc., 2004.

[11] P. Burton, "A Canadian Perspective on the IAEA's State-level Concept," presented at the Symposium on International Safeguards, Vienna, Austria, 2014.

[12] G. G. M. Cojazzi, G. Renda, L. K. Kim, C. Versino, and E. Wolfart, "Acquisition Paths Analysis: Some Methodological Considerations," presented at the Acquisition Path Analysis Workshop, Vienna, Austria, 24-Feb-2014.

[13] G. Renda, L. K. Kim, R. Jungwirth, F. Pabian, E. Wolfart, and G. G. M. Cojazzi, "The Potential of Open Source Information in Supporting Acquisition Pathway Analysis to Design IAEA State Level Approaches," presented at the Symposium on International Safeguards: Linking Strategy, Implementation and People, Vienna, Austria, 2014.

[14] L. K. Kim, G. Renda, and G. G. M. Cojazzi, "Methodological Aspects on the IAEA State Level Concept and Related Acquisition Path Analysis," presented at the 37th ESARDA Annual Meeting, Manchester, UK, 2015.

[15] "Open Source Indicators and Asymmetric Advantage in Security Planning," *International Relations and Security Network*. [Online]. Available: http://www.isn.ethz.ch/Digital-Library/Articles/Detail/?ots591=4888caa0-b3db-1461-98b9-e20e7b9c13d4&lng=en&id=192425. [Accessed: 22-Jul-2015].

[16] Michail Tsikerdekis and Sherali Zeadally, "Online Deception in Social Media," University of Kentucky, 12, Sep. 2014.

[17] Z. Liu and S. Morsy, "Development of the Physical Model," in *Proceedings of the IAEA Safeguards Symposium*, 2007.

[18] D. J. Sweeney and W. S. Charlton, "Proliferation Pathway Decision Analysis for Nuclear Weapons Latency," in *Proceedings of the INMM 52nd Annual Meeting*, Palm Desert, CA, 2011.

[19] D. G. Malcolm, J. H. Roseboom, C. E. Clark, and W. Fazar, "Application of a Technique for Research and Development Program Evaluation," *Operations Research*, vol. 7, no. 5, pp. 646–669, 1959.

[20] H. Thayer, *Management of the Hanford Engineer Works in World War II: How the Corps, DuPont, and the Metallurgical Laboratory Fast Tracked the Original Plutonium Works*. New York, NY, USA: ASCE Publications, 1996.

[21] R. Harney, G. Brown, M. Carlyle, E. Skroch, and K. Wood, "Anatomy of a Project to Produce a First Nuclear Weapon," *Science & Global Security*, vol. 14, no. 2–3, pp. 163–182, 2006.

[22] "Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems (Revision 6)," Generation IV International Forum, GIF/PRPPWG/2011/003, Sep. 2011.

[23] A. Vincze and A. Nemeth, "Effect of State-specific Factors on Acquisition Path Ranking," presented at the IAEA Safeguards Symposium, 2014.

[24] R. S. Kemp, "The Nonproliferation Emperor Has No Clothes," *International Security*, vol. 38, no. 4, pp. 39–78, Apr. 2014.

[25] D. J. Sweeney, J. M. Slanker, W. S. Charlton, and R. Juzaitis, “Quantifying Nuclear Weapons Latency,” in *Proceedings of the 2009 Annual Meeting of the Institute of Nuclear Materials Management*, Tucson, AZ, 2009.

[26] D. J. Sweeney and W. S. Charlton, “Simulating State Proliferation for Nuclear Weapons Latency,” presented at the INMM 54th Annual Meeting, Palm Desert, California, 2013.

[27] E. B. Staats, “Quick and Secret Construction of Plutonium Reprocessing Plants: A Way to Nuclear Weapons Proliferation?,” U.S. General Accounting Office, Washington, DC, Oct. 1978.

[28] V. Gilinsky, M. Miller, and H. Hubbard, “A Fresh Examination of the Proliferation Dangers of Light Water Reactors,” Nonproliferation Policy Education Center, Washington, DC, Oct. 2004.

[29] S. A. Levin, D. E. Hatch, and E. Von Halle, “Production of Enriched Uranium for Nuclear Weapons by Nations X, Y, and Z by Means of the Gas Centrifuge Process,” Union Carbide Nuclear Company, Feb. 1960.

[30] S. Singh and C. R. Way, “The Correlates of Nuclear Proliferation: A Quantitative Test,” *Journal of Conflict Resolution*, pp. 859–885, 2004.

[31] J. Li, M.-S. Yim, and D. N. McNelis, “Model-Based Calculations of the Probability of a Country’s Nuclear Proliferation Decisions,” *Progress in Nuclear Energy*, vol. 52, no. 8, pp. 789–808, Nov. 2010.

[32] J. Ullom, “Enriched Uranium Versus Plutonium: Proliferant Preferences in the Choice of Fissile Material,” *The Nonproliferation Review*, 1994.

[33] A. H. Montgomery and A. Mount, “Misestimation: Explaining US Failures to Predict Nuclear Weapons Programs,” *Intelligence and National Security*, 2014.

[34] R. Kessler and M. Knapik, “Argentine Enrichment Pronouncement Characterized as Startling,” *Nucleonics Week*, Nov. 1983.

[35] D. Albright, “Bomb Potential for South America,” *Bulletin of the Atomic Scientists*, vol. 45, no. 4, p. 16, 1989.

[36] P. Lehner, A. Michelson, and L. Adelman, “Measuring the Forecast Accuracy of Intelligence Products,” The MITRE Corporation, Dec. 2010.

[37] S. Rasmussen, *Production Economics: The Basic Theory of Production Optimisation*, 2nd ed. Springer, 2013.

[38] M. D. Zentner, G. L. Coles, and R. J. Talbert, “Nuclear Proliferation Technology Trends Analysis,” Pacific Northwest National Laboratory, PNNL-14480, Sep. 2005.

[39] J. E. C. Hymans, “Botching the Bomb,” *Foreign Affairs*, Jun-2012.

[40] K. Budlong-Sylvester, J. Pilat, G. Anzelon, C. L. Murphy, C. Reynolds, and B. Boyer, “The Use of Performance Targets in the State-Level Concept,” presented at the 55th Annual INMM Meeting, Atlanta, GA, 2014.

[41] C. Listner, M. J. Canty, A. Rezniczek, and G. Stein, “Approaching Acquisition Path Analysis Formally - A Comparison Between AP and Non-AP States,” presented at the ESARDA Symposium, Bruges, Belgium, 2013.

[42] D. Kahneman and G. Klein, “Conditions for Intuitive Expertise: A Failure to Disagree.,” *American Psychologist*, vol. 64, no. 6, pp. 515–26, Sep. 2009.

[43] “Nuclear Safeguards and the International Atomic Energy Agency,” U.S. Congress, Office of Technology Assessment, Washington, DC, OTA-ISS-615, Jun. 1995.

[44] J. Kun, “Big Data Algorithms Can Discriminate and It’s Not Clear What To Do About It,” *The Conversation*, 13-Aug-2015.

[45] Zubin Jelveh and Michael Luca, “Towards Diagnosing Accuracy Loss in Discrimination-Aware Classification: An Application to Predictive Policing,” presented at the Workshop on Fairness, Accountability, and Transparency in Machine Learning 2015, Lille, France, 2015.

[46] Berkeley J. Dietvorst, Joseph P. Simmons, and Cade Massey, “Algorithm Aversion: People Erroneously Avoid Algorithms After Seeing Them Err,” *Journal of Experimental Psychology: General*, Jul. 2014.

[47] F. L. Greitzer, R. V. Badalamente, and T. S. Stewart, “Collaborative Human-Machine Nuclear Non-Proliferation Analysis,” Pacific Northwest Laboratory, PNL-8922, Oct. 1993.

[48] “The 2014 IAEA Safeguards Implementation Report,” International Atomic Energy Agency, Available at: http://armscontrollaw.com/2015/07/05/2014-iaea-safeguards-implementation-report/, May 2015.

Advancing Disarmament Verification Tools: A Task for Europe?*

Malte Göttsche[1], Moritz Kütt[2], Götz Neuneck[3], Irmgard Niemeyer[4]

[1] University of Hamburg, Germany; current address Princeton University, USA, mgoettsche@princeton.edu
[2] Technische Universität Darmstadt, Germany, kuett@ianus.tu-darmstadt.de
[3] Institute for Peace Research and Security Policy, Germany, neuneck@ifsh.de
[4] Forschungszentrum Jülich GmbH, Germany, i.niemeyer@fz-juelich.de

* This paper was originally published by the EU Non-Proliferation Consortium as Non-Proliferation Paper 47, supported by the EU.

Summary

A number of scientific-technical activities have been carried out to establish more robust and irreversible disarmament verification schemes. Regardless of the actual path towards deeper reductions in nuclear arsenals or their total elimination in the future, disarmament verification will require new verification procedures and techniques. This paper discusses the information that would be required as a basis for building confidence in disarmament, how it could be principally verified and the role Europe could play.

Various ongoing activities are presented that could be brought together to produce a more intensified research and development environment in Europe. The paper argues that if 'effective multilateralism' is the main goal of the European Union's (EU) disarmament policy, EU efforts should be combined and strengthened to create a coordinated multilateral disarmament verification capacity in the EU and other European countries. The paper concludes with several recommendations that would have a significant impact on future developments. Among other things, the paper proposes a one-year review process that should include all relevant European actors. In the long run, an EU Centre for Disarmament Verification could be envisaged to optimize verification needs, technologies and procedures.

I. Introduction

In her statement in the general debate at the ninth Review Conference of the 1968 Treaty on the Non-Proliferation of Nuclear Weapons (Non-Proliferation Treaty, NPT) in 2015, Federica Mogherini, the High Representative of the European Union for Foreign Affairs and Security Policy and Vice-President of the European Commission, reaffirmed the commitment of European Union (EU) member states to pursue nuclear disarmament in accordance with Article VI of the NPT and stressed the need for concrete progress in this field, especially through an overall reduction in the global stockpile of nuclear weapons.[1] In 2010, the eighth NPT Review Conference had reiterated the commitment to nuclear disarmament and the 'total elimination of nuclear weapons', applying 'the principles of irreversibility, verifiability and transparency in relation to the implementation of their treaty obligations'.[2] These important requirements will require well-elaborated, certified and robust technical procedures and technologies.

After the failure of the 2015 NPT Review Conference to reach agreement on a final document, new political initiatives, feasible concepts, courageous actions and technical work will be required to advance nuclear disarmament. Although the momentum for a quick path to a 'world without nuclear weapons', as proposed by US President Barack Obama in 2009, is fading, technical preparations for the verification of nuclear dismantlement and disarmament must continue. The non-approved draft final document of the 2015 NPT Review Conference stated in paragraph 152 that 'The Conference welcomes efforts towards the development of nuclear disarmament verification capabilities that will contribute to providing assurance of compliance with nuclear disarmament agreements for the achievement and maintenance of a nuclear-weapon-free world'.[3]

Although current political progress has been limited, technically oriented preparations can be undertaken now to enable irreversible disarmament when it becomes politically feasible. In particular, 'deep cuts' in nuclear arsenals through 'classical arms control measures' and the prospect of complete nuclear disarmament would require new technical verification measures.[4] Whereas deep cuts require verification of the dismantlement and storage of an agreed number of weapons ('verification of presence'), complete disarmament requires, at the final stage, verification of the fact that no nuclear weapons exist anywhere ('verification of absence'). There is a broad continuum of processes, techniques and technical methods between

[1] See Mogherini, F., EU statement, General Debate, 2015 Review Conference of the Parties to the Treaty on the Non-Proliferation of Nuclear Weapons (NPT), New York, 28 Apr. 2015, <http://www.un.org/en/conf/npt/2015/statements/pdf/EU_en.pdf>.

[2] 2010 Review Conference of the Parties to the Treaty on the Non-Proliferation of Nuclear Weapons, Final document, vol. 1, NPT/CONF.2010/50 (vol. I), New York, 18 June 2010, <http://www.un.org/en/conf/npt/2010/>.

[3] 2015 Review Conference of the Parties to the Treaty on the Non-Proliferation of Nuclear Weapons, Draft final document, NPT/CONF.2015/R.3, New York, 21 May 2015, <http://www.reachingcriticalwill.org/images/documents/Disarmament-fora/npt/revcon2015/documents/DraftFinalDocument.pdf>, p. 17.

[4] For pragmatic recommendations, see Deep Cuts Commission, *Strengthening Stability in Turbulent Times*, Second report of the Deep Cuts Commission (Deep Cuts Commission: Hamburg, Apr. 2015), <www.deepcuts.org/publications/reports>.

both objectives that needs to be elaborated and developed in detail in order to create confidence among all parties involved that the agreed goals of the disarmament regimes have been achieved.

Under the current conditions of an NPT-dominated world, such efforts would also be obligatory for all states parties on the basis of Article VI of the NPT: 'Each of the Parties to the Treaty undertakes to pursue negotiations in good faith on effective measures relating to cessation of the nuclear arms race at an early date and to nuclear disarmament, and on a treaty on general and complete disarmament under strict and effective international control.'[5]

As the NPT demands 'international control', it would seem to be necessary for non-nuclear weapon states (NNWS) to play a more substantive and supportive role in all or at least most verification tasks. Concrete steps towards enabling verification of the disarmament process could be one part of demonstrating compliance with Article VI, bearing in mind the grand bargain of the NPT which requires a commitment to both disarmament and non-proliferation.

There are already a number of verification technologies linked to ensuring non-proliferation and safeguarding fissile materials. A number of NPT Member States, while committed to cooperate with the IAEA in order to equip the IAEA with state-of-the-art verification methods and technologies, have established joint safeguards R&D programmes with the IAEA, the so-called Member State Support Programmes (MSSPs). The Comprehensive Nuclear-Test-Ban Treaty Organization (CTBTO) promotes the development and constant maintenance of technologies to verify the non-existence of nuclear testing. The EU has a high degree of expertise and wide experience in the fields of nuclear safety and security, safeguards and non-proliferation in both nuclear weapon states (NWS) and NNWS.

By contrast, on effective control and monitoring of nuclear disarmament (or 'disarmament verification') there have been only few technological developments, and little has been published in recent decades. In 1967, the Untied States Department of Defense and the US Arms Control and Disarmament Agency (ACDA) conducted 'Field Test FT-34' on developing and testing inspection procedures to monitor the demonstrated dismantlement of nuclear warheads. The test also aimed to evaluate the effectiveness of various evasion techniques, such as diverting fissile material, and to assess the effectiveness of assay operations on fissile material.[6] In 1989, the Natural Resources Defense Council (a US-based non-governmental organization) and a Soviet team from the Russian Academy of Science organized the 'Black Sea Experiments' to determine whether a nuclear warhead was on board a Soviet nuclear-armed cruiser. This was the first time that scientists were allowed to conduct radiation measurements on an operational Soviet nuclear warhead, using a high-resolution germanium detector.[7] The lessons learned and results were published in 1990.[8]

Other activities have been carried out more recently (see table 1). The Trilateral Initiative—a cooperative project between Russia, the USA and the International Atomic Energy Agency (IAEA) in 1996–2002—aimed to establish a verification system under which Russia and the USA might submit excess fissile material to IAEA monitoring.[9] The US-based Nuclear Threat Initiative (NTI) established a verification pilot project in 2012 and published its results in 2014.[10] Interesting projects were also carried out in collaborations between the United Kingdom and Norway as well as between the USA and the UK, all of which raised the need for further research.[11]

More recently, some NWS, in particular the USA, have expressed an interest in and initiated further research on this issue. In 2014, the US State Department launched an 'International Partnership for Nuclear Disarmament Verification' by proposing 'to work with both nuclear weapon states and non-nuclear weapons states to better understand the technical problems of verifying nuclear disarmament, and to develop solutions'.[12] In March 2015 a meeting was held in Washington, DC, in which 26 countries and the EU participated.

In contrast, there have been very few coordinated efforts in Europe, with the exception of Norway and the UK, which have worked together on several exercises simulating the warhead dismantlement process.[13] However, the EU seems destined to play a coordinating role in the emerging sector of irreversible dismantlement verification, due to its non-proliferation and safeguards expertise. NWS and NNWS ought to work together because only a combination of both perspectives can open up avenues for irreversible multilateral nuclear disarmament. Some advances have been made. The European Safeguards Research and Development Association (ESARDA) has added special sessions on disarmament verification to its biannual symposia, although ESARDA generally focuses on improving

5 Treaty on the Non-Proliferation of Nuclear Weapons (Non-Proliferation Treaty, NPT), opened for signature 1 July 1968, entered into force 5 Mar. 1970, <http://disarmament.un.org/treaties/t/npt/text>. See also, e.g. 2015 Review Conference of the Parties to the Treaty on the Non-Proliferation of Nuclear Weapons, 'The United Kingdom–Norway initiative: further research into the verification of nuclear warhead dismantlement', Working Paper submitted by the Norway and the United Kingdom, NPT/CONF.2015/WP.31, 22 Apr. 2015, <http://www.un.org/en/conf/npt/2015/pdf/NPT-CONF2015-WP.31_E.pdf>.

6 See Cliff, D., Elbahtimy, H. and Persbo, A., 'Verifying warhead dismantlement: Past, present, future', VERTIC Research Report no. 9 (Sep. 2010), p. 22.

7 For more detail see Cliff, Elbahtimy and Persbo (note 6), p. 36.

8 Fetter, S. et al., 'The Black Sea experiment', *Science & Global Security*, vol. 1 (1990), pp. 323–33.

9 Shea, T. E. and Rockwood, L., 'Nuclear disarmament: The legacy of the Trilateral Initiative', Deep Cuts Working Paper no. 4 (Mar. 2015), <http://www.deepcuts.org/images/PDF/DeepCuts_WP4_Shea_Rockwood_UK.pdf/>.

10 Hartigan, K., Hinderstein, C. and Newman, A. (eds), *Innovating Verification: New Tools and New Actors to Reduce Nuclear Risks* (Nuclear Threat Initiative: Washington, DC, 2014), <www.nti.org/analysis/reports/innovating-verification-new-tools-new-actors-reduce-nuclear-risks/>.

11 UK–Norway Initiative (note 5); US Department of Energy (DOE), National Nuclear Security Administration, NPAC and British Ministry of Defence, *Joint US–UK Report on Technical Cooperation for Arms Control* (US DOE: Washington, DC, 2015), <http://nnsa.energy.gov/sites/default/files/Joint_USUK_Report_FINAL.PDF>.

12 Gottemoeller, R., Under Secretary of State, Prague, 4 Dec. 2014.

13 UK–Norway Initiative (note 5).

Project	Research areas	Results / remarks
Trilateral Initiative (Russia, USA, IAEA)	Information barrier development, inventory monitoring systems	Cooperation between two nuclear weapon states
United Kingdom—Norway Initiative	Managed access, information barrier development, confidence in verification processes	Cooperation between NWS and NNWS requires more research
Pilot Verification Project (NTI)	Proposals for: baseline declarations, global verification capacity, societal verification	Multinational cooperation required
US–UK cooperation to address technical challenges in verification of nuclear disarmament	Managed access, measurement technologies, information barrier development, chain of custody	Cooperation between two NWS
International Partnership for Nuclear Disarmament Verification (USA)	Only recently announced, practical research areas and activities under discussion	

Table 1. Major projects related to international collaboration on disarmament verification research

international safeguards. Arms control is also a topic in some of the eight ESARDA working groups.[14]

This paper provides information about the current state of research and development (R&D) and outlines new tasks for the European research community. Section II explains disarmament verification in more detail. Section III discusses possible areas for European-wide activity, emphasizing the benefits of increased engagement. Section IV draws conclusions and makes recommendations.

II. Verifying nuclear disarmament and dismantlement

The disarmament process

It is not yet clear which future framework agreements may lead to further nuclear arms control and from there to complete disarmament (i.e. a world without nuclear weapons). It seems clear, however, that this vision can only be achieved through a multilateral process, which includes NWS as well as NNWS. In addition, robust and effective verification is a decisive precondition for maintaining disarmament progress on the way to a world free of nuclear weapons. Regardless of the specific provisions of future regimes, the disarmament and verification process can be discussed from a more technical perspective. One essential task is to identify which verification activities and combinations thereof are suited to providing confidence in irreversible nuclear disarmament. One way of approaching this is to envisage a world without nuclear weapons and deliberating on which verification activities should have been in place during the disarmament process to obtain confidence in the absence of nuclear warheads. From a logical point of view, confidence would be needed that (*a*) no nuclear warheads exist and (*b*) no fissile material can be used to build nuclear warheads anymore.

In order to achieve (*a*) and (*b*), certain information must be available, such as the total number of existing nuclear warheads and the number being dismantled, furthermore the quantity of remaining fissile material would have to be known, and this would need to include knowledge about the quantities being disposed of and newly produced. This information would be required in order to make a record of all existing fissile material and to be able to ensure that no fissile material remained outside the verification regime.[15] All the information provided would need to be strictly and effectively evaluated, as confidence in non-verified declarations would remain very limited. Information on existing warheads and materials can be verified for their correctness. Further measures would be necessary, however, to verify that the information declared was complete and that no hidden stocks of warheads and fissile materials existed.

So-called baseline declarations of total numbers of delivery vehicles, warheads and materials stocks would indicate current arsenals and capabilities.[16] Certain metadata would have to be part of such declarations in order to enable verification measures. This includes the location of items and materials or a subdivision of items and materials into certain categories, such as warhead types or the purpose of materials. Once baseline declarations have been issued, changes in arsenals and stocks would need to be declared. By comparing change declarations with the baseline, the total inventory would become known over time, assuming there were no undeclared stocks (see figure 1).

Change declarations would include information regarding the physical dismantlement of warheads or delivery vehicles, the disposition of materials and the production of new materials and warheads. The disassembly of a warhead must be handled in a protected environment or in special disassembly facilities with controlled access, due to the need for safety and security arrangements. The

[14] For more information see the ESARDA website, <http://www.esarda.eu/>.

[15] This paper only focuses on military fissile material stockpiles. Civilian stockpiles present additional challenges, although there is considerable experience of dealing with these within the IAEA and Euratom.

[16] Fuller, J. et al.,'Verifying baseline declarations of nuclear warheads and materials', eds Hartigan, Hinderstein and Newman (note 10).

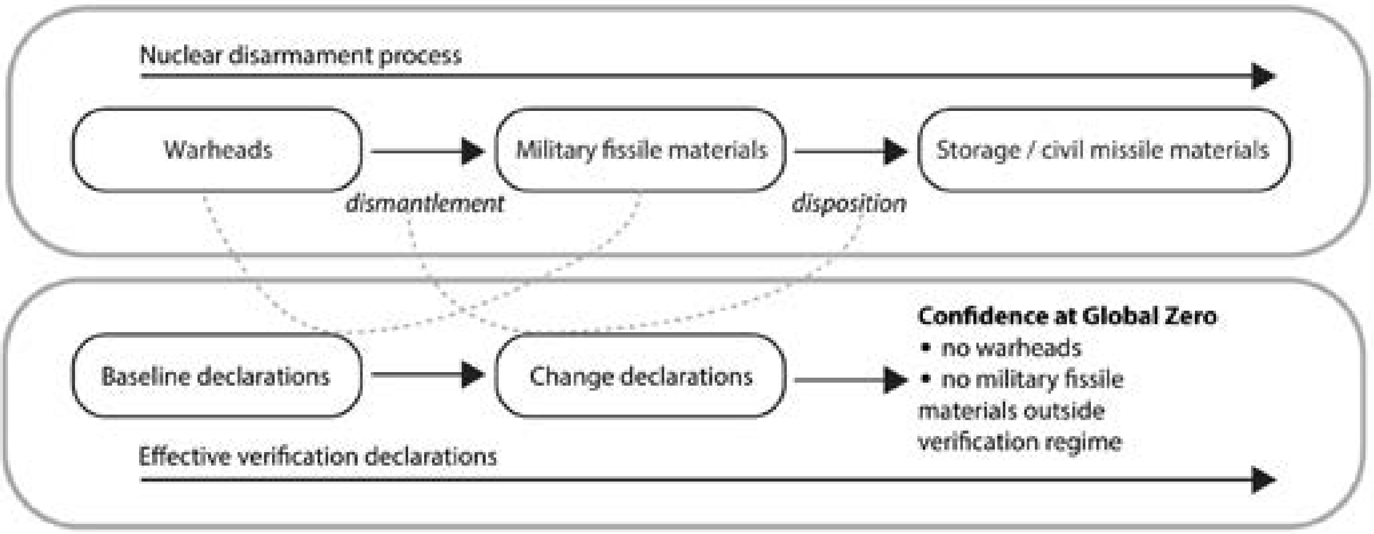


Figure 1: The technical disarmament process (above) and declarations (below) requiring effective verification to enable future confidence in a world without nuclear weapons (Global Zero). The dashed lines show which declarations are relevant for specific steps in the disarmament process. Please note that the timelines of both processes are different.

three-stages of the dismantlement process are (*a*) the warhead is placed in a disassembly facility and is identified; (*b*) warhead disassembly (of high-explosives, fissile material and non-nuclear components); and (*c*) component disposition and storage in the same or a different location.[17] Verification must provide confidence in this process. Technical visits to nuclear storage facilities by inspectors would be needed to verify the stored warheads and whether warheads have been transported to other locations. In addition, verification is required of the removal of warheads from delivery vehicles and other deployment-status changes.

Logically, confidence in the absence of nuclear warheads could be gained even though weapons-usable fissile materials remain, as long as their non-diversion has been verified. Irreversibility would be maximized, however, if these materials were disposed. Materials could either be converted to civilian or commercial purposes (both uranium and plutonium can be used as reactor fuel) or transported to long-term storage. At this point, the particular disarmament verification issues linked to the protection of sensitive information will no longer exist and verification activities will be similar or identical to non-proliferation safeguards. In the very long term, declarations could possibly indicate that no directly weapon-usable fissile material exists.

Measures are needed to verify both the correctness and the completeness of the information declared by states. Analysis of the numbers provided by baseline and change declarations is suitable for verifying the correctness of declarations, but additional measures might be necessary to verify the absence of undeclared stocks of warheads and fissile materials—the completeness of declarations. Further declarations of data and information as well as a willingness to deal with requests for additional information will be instrumental to enabling at least a certain degree of confidence in completeness. This may include, among other things, details of past fissile material production to determine whether this data is consistent with the declared stocks and information on further facilities in cases of suspicious activity. Confidence in the absence of warheads can be increased by evaluating information on weapon delivery systems, military force structures and military doctrines for their consistency with declarations on disarmament.

Building confidence in the correctness of warhead and fissile material stock declarations and in the irreversible dismantlement and disposition processes will require time. During this long process, declarations would be checked periodically for consistency. It will be a much bigger challenge to successfully and consistently cheat over a long period, throughout years of verification activities, compared to just a single false declaration. Accordingly, confidence in the correctness and completeness of declarations will increase over time if no inconsistencies are discovered. Hiding stocks or material production facilities from declaration will be a more difficult undertaking over a period of decades, especially if the total amount of fissile material decreases. Verification of warhead and material stocks should therefore ideally commence as soon as possible. Such declarations and verification measures could occur before any disarmament activities take place.

[17] See Cliff, Elbahtimy and Persbo (note 6).

Verification techniques and procedures

Although IAEA safeguards have many overlaps with disarmament verification requirements, such as the verification of nuclear material storage, verification of the basic disarmament process described above will require somewhat different techniques and procedures in order to meet the special challenges associated with disarmament verification. Accordingly, further technical research, development and testing will be needed to adapt existing technologies to such needs.

Three major, unprecedented challenges for disarmament verification have to be met. First, the most relevant information cannot be shared by the inspected state. Articles I and II of the NPT prohibit NWS from transferring proliferative knowledge such as warhead design properties to NNWS, and vice-versa. Other information might be retained for national security reasons, without specifying why. Thus, access to information by inspectors from NWS might be severely restricted. Inspectors' access to data will be strictly controlled. During on-site inspections, inspector movement inside sensitive facilities would be extremely limited—and many items could be masked. Moreover, the direct measurement of sensitive items would not be possible as the results would contain sensitive information. All this would create major challenges for the inspectors. It would be essential to work out procedures to address these challenges.

Second, safety and security requirements will need to be observed and could, under no circumstances, be compromised. In addition to access by personnel, tools must be certified for use near or on a nuclear explosive device, in particular with regard to explosive safety.[18] In some cases, special verification tools must be developed that could pass safety and security evaluations. The specifics of these evaluations are often sensitive, which requires careful cooperation between inspecting and inspected state or the IAEA.

Third, a particularly high level of confidence would be required in all verification activities. Verification failures concerning just a single warhead would not be acceptable. Some of the techniques applied in IAEA safeguards might not meet such high standards and, therefore, could not be used in disarmament verification. It follows that such procedures must be tested in advance and be developed further. A determined and comprehensive effort will be required on the development of new tools, techniques and procedures that can provide a sufficiently high level of confidence while at the same time complying with all restrictions.

Four generic types of methods are relevant to verifying warhead and fissile material stocks and processes: (*a*) authentication; (*b*) unique identification; (*c*) continuity of knowledge, to verify the correctness of declarations; and (*d*) nuclear archaeology, among other things, to verify their completeness.

Methods of *authentication* verify that an item truly is what it is declared to be. It is essential to ensure that declared warheads have not been replaced by mock-ups or fake warheads. Mock-ups could, for example, be used to divert declared items and materials in order to build up a hidden stock. Authentication poses immense challenges for sensitive items and materials where measurement data cannot be directly disclosed because it would contain sensitive information. This is the case not only for warheads, but also for the fissile materials that were produced for warheads or the result of warhead dismantlement. Typically, two authentication methods can be identified: the template approach and the attribute approach.

The template approach is based on the comparison of a reference item (the 'golden sample') with a test item (i.e. the warheads and materials to be authenticated). If test and reference items are equal and the golden sample is the declared warhead or fissile material it is supposed to be, then the test item is successfully authenticated. The main problems are the selection of the golden sample and of the initial test that it is indeed the presumed object. The attribute approach is based on measurements of several predefined parameters of a warhead, for example, the presence of specific materials, their composition and their mass. Ranges of possible values can be defined for each attribute based on the certified uncertainty of measurement and publicly known specifications of nuclear warheads. If all the attributes of an item are within this range, the item is successfully authenticated. The attribute types and quantitative ranges can be agreed among the inspected and inspecting states during verification procedure negotiations.

Both approaches rely on numerous nuclear measurement techniques (see table 2). The main measurement methods are passive gamma (g) spectrometry and neutron multiplicity measurements.[19] Both are non-destructive assay techniques that leave the samples intact. They can be used to estimate isotopic compositions, plutonium presence and the mass of the item. To determine uranium presence and mass, however, active interrogation methods must be used. These are based on neutron interrogation and measuring the resulting prompt/delayed neutrons and gammas, as well as a variety of imaging techniques based on g-rays. They also include nuclear resonance fluorescence methods.[20] Some of these techniques are already used for IAEA safeguards, but none of them are reliable enough yet to be used for authentication. While the item configuration (e.g. its geome-

[18] Fuller et al. (note 16), p. 24.

[19] Gamma spectrometry records the energy spectrum of gamma rays emitted due to the radioactive decay of different sources and can be used to identify materials and material compositions. Neutron multiplicity measurements are based on correlations between the neutrons emitted by the radioactive decay and the reactions of materials, and can be used to analyse the mass of a neutron emitting material in a sample. Nuclear resonance fluorescence can be used to identify isotopes using fluorescence effects after a material has been irradiated with high-energy photons. See Göttsche, M. and Kirchner, G., 'Measurement techniques for warhead authentication with attributes: advantages and limitations', *Science & Global Security*, vol. 22, no. 2 (2014), pp. 83–110.

[20] Nuclear resonance fluorescence can be used to identify isotopes using fluorescence effects after a material has been irradiated with energy photons.

try) is usually known in IAEA safeguards, little or no useful information is available for disarmament verification purposes. The measurement techniques are, in particular, vulnerable to the presence of further materials that shield radiation emitted from the fissile material, such as explosives or the safety storage containers that hold warheads and fissile materials. Such materials might lead to incorrect results, such as wrong mass estimates. The high level of reliability required makes further development of measurement techniques necessary.[21] Such R&D should be directed at reducing the influence of shielding and other effects of an unknown configuration on the measurement results.

Different solutions have been proposed to protect sensitive information. So-called information barriers are technical devices that carry out detailed measurements but, instead of revealing detailed measurement data, only show the verification result. For instance, the device might give a green light if a warhead is identified; a red light if it is a different item; or a yellow light if the measurement is inconclusive. In addition to measurement reliability, information barriers present further challenges. The inspected state must be able to ensure that the inspecting state (or the IAEA) has not secretly built in a capacity to leak sensitive information. This could be achieved by sophisticated equipment certification prior to the measurements being taken. Both the inspecting state and the inspected state must have confidence in the authenticity of the equipment, for instance, that the information barrier was not tampered with by modifying the analysis algorithm in order to give false results. Several multinational initiatives, such as the Trilateral and UK–Norway initiatives, have attempted to build such information barriers, but all of them either only covered a limited number of attributes or were not fully trusted, authenticated and certified by all parties.

Such equipment authentication and certification challenges could be reduced by implementing a ‘zero-knowledge protocol’, which does not require an information barrier. This approach avoids releasing sensitive information by specifically designed test procedures that apply non-electronic differential measurements and never measure sensitive information.[22] This technique, however, also requires further research, development and testing of equipment.

After authentication, the tested item or material container should be given a *unique identification*. This allows inspectors to recheck an item or the container again later without the need for new measurements. This is especially important as items may change locations. Without unique identification, inspectors would risk double counting items. If an item is missing, the inspectors will know exactly which item it is. Items can be uniquely identified by attaching a tag. This needs to be very robust to withstand tampering and ensure the safety of the tagged item, taking explosives safety into account.[23] Inspected states would have the capacity to invest vast amounts of resources in defeating tags, so the development of tamper-proof tags would be required, including the tools to identify tampering in a timely manner. Appropriate tags have not yet been successfully developed. Promising approaches and methods do exist (see table 2), but more testing and development are required.[24]

Unique identifiers would also help to preserve the *continuity of knowledge*, by which inspectors will be able to follow items through time and processes. Achieving continuity of

Verification type	Examples of verification tools and techniques
Warhead and military fissile materials authentication by attribute and template systems	Gamma spectrometry
	Passive neutron multiplicity counting
	Active methods, such as neutron interrogation, neutron imaging and nuclear resonance fluorescence
	Zero-knowledge protocol systems
Unique identification	Ultrasonic intrinsic tag
	RuBee tag
Continuity of knowledge	Unique identification tools
	Unattended monitoring, for example, using cameras and sensors
	Seals
	Managed access
Measures to verify the completeness of declarations	Nuclear archaeology (past fissile materials production)
	Challenge inspections at suspicious sites
	Open source and intelligence data analyses
	Satellite imagery

Table 2: Verification tools and techniques that require further research, development and testing

[21] Fuller et al. (note 16), p. 33.

[22] Glaser, A., Barak, B. and Goldston, R. J., ‘A zero-knowledge protocol for nuclear warhead verification’, *Nature*, 26 June 2014, pp. 497–502.

[23] These are classic tasks for safeguards and have already been applied, but current technologies must be developed further.

[24] Fuller et al. (note 16), p. 36.

knowledge is a significant challenge because access to facilities by inspectors and the breadth of allowable inspection measures will be severely restricted by the inspected party for reasons of non-proliferation and national security. Continuity of knowledge is of particular importance during warhead dismantlement. As inspectors may not be able to visually observe the physical dismantlement, procedures and tools based on observations and measurements before and after the activity, while using severely restricted information during the activity itself, must be in place to provide confidence in the dismantlement for the inspectors. The tools and techniques that contribute to the continuity of knowledge are also shown in table 2. Again, they will need to be certified and authenticated. Effective managed access procedures and protocols would restrict access in a reasonable way while at the same time supporting the continuity of knowledge by providing inspectors with sufficient access to the facility but preventing the disclosure of sensitive information. Past projects have shown that strict enforcement of access controls by the inspected party can negatively affect the confidence of inspectors in the process.[25] Further exercises should be conducted to identify procedures and protocols that balance safety, security and the protection of sensitive information, on the one hand, with enabling inspector confidence, on the other.

Verification of the completeness of a declaration requires additional tools and techniques. Like the safeguards under an IAEA Additional Protocol, challenge inspections could be conducted at suspicious sites, but access to facilities might remain severely limited if sensitive information were at stake, and as a result the confidence gained from such activities may remain low. Open source and intelligence information, including satellite imagery, can also be analysed in connection with suspicious sites and activities, but the level of confidence obtained from the use of these techniques may not be sufficient. Comparing past fissile material production to declared warhead and materials stocks can be a very powerful tool. Past production can be estimated by applying nuclear forensic methods or, more specifically, *nuclear archaeology* could be used. If the declared warhead and fissile material stocks are in agreement with the estimated material production, confidence in the absence of undeclared materials or warheads is increased. Nuclear archaeology techniques analyse microparticles and the activation products found at fissile material production facilities to estimate their production histories over long periods of time.[26] It is, however, necessary to carry out such measurements before plants are decommissioned and, like the other methods, more research in sampling technologies is needed to achieve a high degree of accuracy and validity of results. Overall, verification of the absence of hidden stocks may pose the largest verification challenge and will require substantive R&D.

[25] UK–Norway Initiative (note 5), p. 8.

[26] Fetter, S., 'Nuclear archaeology: verifying declarations of fissile material production', *Science & Global Security* vol. 3, nos. 3–4 (1993).

Finally, measures on nuclear disarmament would be strongly supported by the successful negotiation of a treaty banning the production of fissile material for nuclear weapons or other nuclear explosive devices (a Fissile material cut-off treaty, FMCT). Such a treaty would entail further verification challenges, the most obvious of which are ensuring that there is no (*a*) re-commissioning of shut down enrichment and reprocessing facilities; (*b*) diversion of fissile materials produced at currently operational enrichment and reprocessing plants; (*c*) production at clandestine facilities; (*d*) production at suspect military nuclear facilities; and (*e*) diversion of HEU from naval fuel.[27] The R&D activities required for an FMCT, such as verifying inventories of weapon-usable materials, overlap significantly with warhead dismantlement R&D.

Lessons learned from the verification regimes of other non-proliferation and disarmament treaties

R&D on verification methods and techniques has always played an important role in negotiating and implementing verification regimes for non-proliferation and disarmament treaties. Such R&D activities and their interplay with treaty development and implementation are described below.

The nuclear safeguards system was laid down in the early IAEA safeguards agreements (INFCIRC/26, INFCIRC/66 and INFCIRC/153). It was strengthened based on the lessons learned from the discovery of clandestine nuclear weapon development in Iraq in the early 1990s, the experience gained in verifying South Africa's dismantlement of its nuclear weapon programme, and the verification challenges encountered in the Democratic People's Republic of Korea (DPRK). As a result, the IAEA approved the so-called 93+2 programme, which was in its first part an R&D programme that aimed to identify and adopt additional or new methods and techniques to better support the detection of undeclared activities. The scientific and engineering communities obviously played essential roles in this process. In the second part of the programme, some of the new methods and techniques were incorporated into the IAEA's toolbox as part of the Additional Protocol to current safeguards agreements.

Since 1977, the IAEA has based its technical and scientific programme for nuclear verification on voluntary contributions by member states. These contributions, referred to as the IAEA Member State Support Programmes (MSSPs), have consisted of financial support, R&D, training and consultancy to improve the implementation of safeguards. They, therefore, represent an important pillar in enhancing verification methods and techniques for safeguards purposes and, from a broader view, for non-proliferation regimes in general.

[27] See Schaper, A., 'Verifiying the nonproduction and elimination of fissile material for weapons', ed. C. Hinderstein, *Cultivating Confidence: Verification, Monitoring and Enforcement for a World Free of Nuclear Weapons* (Nuclear Threat Initiative: Washington, DC, 2010), pp. 67–122; and Feiveson, H. et al., *Unmaking the Bomb: A Fissile Material Approach to Nuclear Disarmament and Nonproliferation* (MIT Press: Cambridge, MA, 2014), p. 148.

Recognizing the need to constantly scan the horizon, the IAEA has recently established a technology foresight process focused on instrumentation that could be applicable to safeguards fields and laboratory activities. Again, the involvement and engagement of the scientific and engineering communities have been essential in this regard.

R&D to establish verification methods and technologies for the Comprehensive Nuclear-Test-Ban Treaty (CTBT) also has a long tradition, and has had a significant impact on CTBT negotiation and adoption. The need to develop a scientific basis for monitoring nuclear testing in all environments was explicitly recognized following the first CTBT negotiations in 1958. This marked the start of programmes of basic and applied research that have continued to this day.[28] Within the framework of other arms control treaties, such as the 1993 Chemical Weapon Convention (CWC) and the 1990 Treaty of Conventional Armed Forces in Europe (CFE), sophisticated inspection procedures, such as 'on-site inspections', have been developed and used. Managed access to military facilities is always an enormous challenge for the inspectors and the inspected party alike. This includes the certification of authorized technical equipment, agreed timelines and inspection rights.

During the cold war, however, scientific and technical disagreements over verification methods hindered the successful conclusion of a CTBT. It was in the early 1970s, when a long-term group of scientific experts (GSE) including representatives from up to 40 countries was established to study the technical aspects of monitoring for nuclear explosions, that the scientific and engineering communities started to contribute significantly to the design and implementation of monitoring systems, confidence building, and finally to the successful negotiation of a verification regime and its measures.[29] After the CTBT was opened for signature, the Preparatory Commission for the CTBTO was established in 1996. Since then, the Provisional Technical Secretariat (PTS) has been tasked with establishing a verification regime. The scientific and engineering communities have continued to contribute to the further development and implementation of the International Monitoring System (IMS), the International Data Centre, and procedures and techniques for on-site inspections. Numerous publications and reports, biannual 'science and technology conferences' with more than 1100 registered participants and 550 abstracts in 2015 alone, as well as Integrated Field Exercises aimed at simulating on-site inspections all demonstrate the major contribution of R&D to CTBT verification.

Given the useful role that scientific experts have played in various disarmament negotiations, Germany and the Netherlands held two Scientific Experts Meetings on Technical Issues Related to an FMCT in 2012.[30] About 100 participants attended from 47 states. The first meeting addressed the questions of how facilities for the production of fissile material for nuclear weapons could be decommissioned in a verifiable and transparent manner; how to deal with facilities in nuclear weapon states that were not originally designed with safeguards in mind; and how to handle the transformation of military facilities into civilian facilities. The second meeting focused on the role and limitations of 'nuclear archaeology' in the verification of a future FMCT, with special attention to the detection of secret and/or undeclared activities; and on an FMCT-specific system of managed access and other verification provisions to ensure the non-diversion of nuclear material for prohibited purposes. Further contributions by the scientific and engineering communities are needed in order to move the FMCT negotiations forward. Following four meetings of the Group of Governmental Experts in 2014 and 2015 to discuss recommendations for advancing FMCT negotiations, France announced its intention to issue an initial draft of an FMCT treaty.[31]

All these examples illustrate that coordinated technical discussions are an important element of treaty negotiation and treaty implementation. These important precedents show that technical R&D on disarmament verification capabilities, according to the technical requirements presented above, will be prerequisites for the negotiation of disarmament agreements, and that technical activities can enhance the implementation of disarmament activities.

III. European activities: engagement and its future benefits

This section argues why it should be in the interest of European countries and the EU to play a coordinating role in the emerging sector of irreversible dismantlement verification. Existing European activities and approaches are discussed; and the case is put for multilateralism and participation by the NNWS. The standpoint of the EU on disarmament verification, as a supporter of effective multilateralism, is presented; and the benefits of European engagement articulated.

[28] National Academy of Sciences, *Technical Issues Related to the Comprehensive Nuclear Test Ban Treaty* (National Academies Press: Atlanta, GA, 2002), <http://www.nap.edu/catalog/10471/technical-issues-related-to-the-comprehensive-nuclear-test-ban-treaty\>.

[29] Dahlman, O., Mykkeltveit, S. and Hein, H., *Nuclear Test Ban: Converting Political Visions to Reality* (Springer: Dordrecht, 2009).

[30] Conference on Disarmament, Germany–Netherlands FMCT Scientific Experts Meeting: Technical Issues Related to a Fissile Material Cut-Off Treaty (FMCT), CD/1935, 2012, <http://daccess-dds-ny.un.org/doc/UNDOC/GEN/G12/613/90/PDF/G1261390.pdf?OpenElement?\>; and Conference on Disarmament, Scientific Experts Meeting on technical issues related to a treaty banning the production of fissile material for nuclear weapons or other nuclear explosive devices based on resolution 66/44 of the General Assembly of the United Nations, CD/1943, 2012, <http://daccess-dds-ny.un.org/doc/UNDOC/GEN/G12/626/21/PDF/G1262621.pdf?OpenElement\>.

[31] Conference on Disarmament, Draft Treaty Banning the Production of Fissile Material for Nuclear Weapons or Other Nuclear Explosive Devices, CD/2020, 2015, <http://daccess-dds-ny.un.org/doc/UNDOC/GEN/G15/076/39/PDF/G1507639.pdf?OpenElement\>.

Current European activities and approaches

In her keynote speech at the 2007 Carnegie Conference, the British Foreign Secretary, Margaret Beckett, proposed that the UK should be at the forefront of the conceptual and practical work required to achieve a world without nuclear weapons. She also suggested that the country should become a 'disarmament laboratory' and elaborated concrete steps towards multilateral disarmament. The UK–Norway initiative was established in 2007 to explore effective verification measures, which are an important precondition for fulfilling Article VI of the NPT. At the 2010 NPT Review Conference, the British Government distributed a document, *The Road to 2010*, which proposed the establishment of a 'Nuclear Centre of Excellence' with a planned budget of 20 million pounds (~23.5 million EUR in May 2010), but this initiative was later dropped after a change of government.

In April 2014, British Pugwash discussed a proposal to establish a national 'British International Nuclear Disarmament Institute' (BRINDI), which aimed 'to facilitate the achievement of the complete, stable, sustainable and irreversible elimination of nuclear weapons by creating the enabling conditions towards universal nuclear disarmament'.[32] The independent trilateral Deep Cuts Commission issued a report in May 2015 in which its 21 commissioners from Germany, Russia and the USA recommended, among other things, 'the creation of an international center for nuclear disarmament, research, development, testing and demonstration of fissile material'.[33] All these activities, concepts and proposals underline that the time is ripe to elaborate more detailed R&D programs in order to establish effective multilateral verification measures. First, however, it is useful to describe some concrete past activities involving mainly European actors.

The UK–Norway Initiative

Although not 'intra-EU', the initiative between the UK and Norway is probably the most prominent example of good cooperation in Europe. Since 2009, the two states have cooperated in a broad project. Several organizations have been involved, such as the Norwegian Foreign Ministry and the British Ministry of Defence, the British Atomic Weapons Establishment, several Norwegian research institutions and, at the beginning of the project, the Verification Research, Training and Information Centre (VERTIC). The initiative's central aim has been the simulation of nuclear disarmament verification exercises between a nuclear weapon state and a non-nuclear weapon state. The exercises have been carried out in the form of role plays. One of the goals was to learn about the possible effects of restricted access on inspector–host interaction. A second, more technical strand involved the development, fabrication and testing of an information barrier, which was used during the exercises. The two states have made presentations on outcomes on numerous occasions, including at the most recent meetings of the NPT Review Cycle. The project is currently ongoing. Working with the King's College, London, the project has been extended as a social science study. The verification exercise has been carried out repeatedly in order to gain empirical data on the development of trust and confidence between inspector and host personnel.

ESARDA activities

The European Safeguards Research and Development Association (ESARDA) is an association of European organizations actively involved in R&D on nuclear safeguards. The control of civil nuclear material is mandatory on the territory of EU member states, in line with the 1958 Treaty establishing the European Atomic Energy Community (Euratom Treaty) and the NPT. ESARDA was formed in 1969 to facilitate collaboration on R&D in the field of safeguards and the application of such knowledge to the safeguarding of source and special fissile materials. ESARDA has 32 member organizations and 5 associated partners from Norway, Switzerland and the USA. These include regulatory authorities, operators of nuclear facilities, research centres and universities. The principal areas of activity are the coordination of research, frequent exchanges of information and joint execution of R&D programmes. ESARDA also strives to play an educational role that reaches the general public. To this end, there are (*a*) annual meetings and symposia, which provide opportunities for collaboration and exchanges of scientific information; (*b*) dedicated working group activities, currently by nine working groups; (*c*) a one-week ESARDA Course, which complements nuclear engineering studies by including nuclear safeguards in the academic curriculum; (*d*) the peer-reviewed journal, *ESARDA Bulletin*; and (*e*) the ESARDA website.

ESARDA is currently more active than ever, due to the lively cooperation among its members and its strong linkages to other safeguards-related organizations, such as the US-based Institute of Nuclear Materials Management (INMM), as well as the proactive tackling of new and emerging issues through its diverse working groups. As a result, ESARDA has put more emphasis on specific topics including arms control and disarmament verification, in addition to the traditional safeguards-related topics of ESARDA. A sub-group on arms control verification has been established by the novel approaches/novel technologies working group, and the verification technologies and methodologies working group regularly considers arms control and disarmament verification approaches at its meetings. During the 35th ESARDA Symposium in 2013, the first special panel discussion was held on this very topic. The panellists concluded that ESARDA would be an excellent forum for further deliberations on dismantlement verification, as its members have the required

[32] For more detail see 'Realising the Disarmament Institute (BRINDI)', British Pugwash, 19 Feb. 2015, <http://britishpugwash.org/realising-the-disarmament-institute-brindi/>.

[33] Deep Cuts Commission (note 4).

expertise and ESARDA offers a heterogeneous platform for both nuclear weapon and non-nuclear weapon states.[34]

Nuclear Disarmament Verification Network in Germany

For several years, individual researchers in Germany have met to discuss the issue of nuclear disarmament verification on an independent basis. To clarify their goal, they founded the Nuclear Disarmament Verification Network. Its biannual meetings discuss the progress of research in Germany. The group organized a special session and a panel at the ESARDA Symposium in 2013 to discuss the needs of nuclear disarmament verification. Several institutions are working on disarmament verification in Germany. There are research groups at Forschungszentrum Jülich, the Fraunhofer Institute for Technological Trend Analysis in Euskirchen and the Carl Friedrich von Weizsäcker Centre at the University Hamburg. In close cooperation with the Institute for Peace Research and Security at the University of Hamburg, the Centre is conducting a project on disarmament verification, focused on nuclear weapons authentication using gamma and neutron measurements. Various scientific publications have resulted from a first PhD project that was completed in 2015.[35]

Discussion on the humanitarian consequences of nuclear weapons

EU member states, in particular Austria and Ireland, play a visible role in the global initiative on the humanitarian consequences of nuclear weapons. A discourse about the humanitarian dimension of nuclear weapons and the risks associated with their use has been held for several years as part of international negotiations. The humanitarian consequences or impact of the use of these weapons was addressed by several states and debated in three international conferences in Norway (2013), Mexico (2014) and Austria (2014). While there is no direct relation to technical disarmament verification, it is important to note that calls were made during these conferences for a treaty banning nuclear weapons. Initiated by non-governmental organizations (NGOs), the call was later supported by many countries. The so-called Austrian Pledge for a global legal instrument on the prohibition and elimination of nuclear weapons—later known as the 'Humanitarian Pledge' during the ninth NPT Review Conference—has been supported by 112 states to date. The pledge focuses very much on the use and risk of nuclear weapons, but it does not articulate the huge technical, legal and safety challenges linked to dismantling and destroying existing nuclear weapons. Although the need for verification is not mentioned in the Humanitarian Pledge, it is obvious that more rapid and drastic disarmament would create the need for effective verification measures and instruments, including multilateral verification of nuclear warhead dismantlement.

The case for multilateralism and NNWS participation

If disarmament is to be verified by a multilateral regime and include NNWS, all the inspectors involved will need to have confidence in the functionality of the equipment, as detailed above. If not, the verification measures will not give inspecting states the required confidence in the compliance of inspected states. There is a parallel need for the inspected state to have confidence that all the equipment and procedures will not disclose sensitive information.

Perhaps the only option to achieve this confidence is to directly involve all relevant parties, both NWS and NNWS, in the R&D of all verification measures and related equipment. This would include, among other things, a careful dialogue among the participants about their different expectations and requirements. Attention must be paid to the potential vulnerabilities and limitations of equipment, to enable a comprehensive assessment of its reliability and thus the confidence that can be gained. Assessments could be based on joint testing exercises with the benefit of the ability to discuss all vulnerabilities and limitations and arrive at a mutual understanding.

The need for capacity building and the multilateral engagement of a comprehensive variety of actors was recognized in the Final Document of the 2010 NPT Review Conference: 'All States agree on the importance of supporting cooperation among Governments, the United Nations, other international and regional organizations and civil society aimed at increasing confidence, improving transparency and developing efficient verification capabilities related to nuclear disarmament'.[36]

The European Union: a strong supporter of 'effective multilateralism'?

The 28 member states of the EU hold fairly diverse positions on their approach to nuclear matters. The majority of member states are also members of the North Atlantic Treaty Organization (NATO), and NATO declares itself to be a nuclear alliance for as long as nuclear weapons exist. The countries represent a wide spectrum of nuclear policies, which stem from their different political, geostrategic and cultural origins. France and the UK, two of five recognized NWS, possess small nuclear stockpiles and a mostly sea-based nuclear arsenal. In addition, five NATO members—Belgium, Germany, Italy, the Netherlands and Turkey—host approximately 200 US sub-strategic nuclear weapons in several locations. Non-NATO EU member states such as Finland and Sweden are more committed to a strict disarmament policy. Only Austria, Cyprus, Ireland and Malta have signed the Humanitarian Pledge, but 20 EU member states—including the Baltic

[34] Göttsche, M. and Neuneck, G., 'Panel discussion: disarmament verification, a dialogue on technical and transparency issues', *Esarda Bulletin*, no. 50 (Dec. 2013).

[35] See Göttsche, M. and Kirchner, G., 'Improving neutron multiplicity counting for the spatial dependence of multiplication: results for spherical plutonium samples', *Nuclear Instruments and Methods in Physics Research A* 798, 99–106 (2015); Göttsche, M., 'Reducing neutron multiplicity counting bias for plutonium warhead authentication', PhD dissertation, University of Hamburg, 2015, <http://ediss.sub.uni-hamburg.de/volltexte/2015/7356>; Göttsche and Kirchner (note 19)

[36] 2010 Review Conference of the Parties to the Treaty on the Non-Proliferation of Nuclear Weapons (note 2), p. 24.

states, the Benelux countries, many states of Central and Eastern Europe, as well as Greece, Italy, Germany and Spain—have associated themselves with the 'Statement on the Humanitarian Consequences of Nuclear Weapons' presented by Australia to the NPT Review Conference in 2015. This stated that these countries would 'welcome initiatives to develop a better understanding of the complexities of international nuclear disarmament verification'.[37]

The EU is a strong supporter of the NPT as 'the cornerstone of the global non-proliferation regime' and 'as a key priority, and as a multilateral instrument for reinforcing international peace, security and stability'.[38] The EU has played an important role in the 'EU 3 plus 3 talks' with Iran and helped to agree the key parameters of the Joint Comprehensive Action Plan in 2014.[39]

At the ninth NPT Review Conference, the EU committed 'to continue to promote a comprehensive, balanced and substantive full implementation of the 2010 NPT Action Plan, which includes the total elimination of nuclear arsenals (Action 3) and 'rapidly moving towards overall reductions' (Action 5). Action 19 from 2010 in particular states that: 'All States agree on the importance of supporting cooperation among Governments, the United Nations, other international and regional organizations and civil society aimed at increasing confidence, improving transparency and developing efficient verification capabilities related to nuclear disarmament'.[40]

The EU participated in the ninth Review Conference of the NPT in 2015 and published several useful working papers: on the EU's support for the CTBT (Working Paper no. 50); 'safeguards implementation in the EU' (no. 55) and 'nuclear safety' (no. 56). These reflect the areas of activity of the EU. Despite these efforts, however, the EU had a fairly low profile in terms of activities and action at the conference, even though the EU hosts much experience and research activity and many institutions with related expertise in the field of nuclear verification. From a technical point of view, there are also activities and experience in the field of the security and physical protection of nuclear materials and facilities (Action 40). Under Action 42, the final document called on states 'to improve their national capabilities to detect, deter and disrupt illicit trafficking in nuclear materials'. Special detection equipment for portal monitoring would be required to achieve this. In addition to being used for security, this could also contribute to fissile material control. In Main Committee I (on disarmament), the EU Special Envoy for Non-proliferation and Disarmament, Jacek Bylica, stated that: 'The EU welcomes and encourages the holding of further P5 Conferences on the follow-up to the 2010 NPT Review Conference, including confidence-building, transparency, verification activities and discussions on reporting'.[41]

All these examples illustrate European interest in disarmament verification in principle. European engagement could take different forms. In addition to independent research activities, it could involve cooperation among various research institutions in European countries. Funding could be provided by research councils and other foundations. The activities of ESARDA could be expanded. On a different scale, activities could be initiated at the governmental level and follow the example of the UK–Norway Initiative. European cooperation could then involve cooperation among two or more states. Alternatively, the EU could involve the Joint Research Centre (JRC) and EURATOM to further engage with the issue. The JRC supports the European Commission's work on nuclear safeguards and security, led by the Commission's Directorates-General for Energy and Home Affairs, respectively, by developing efficient and effective systems for safeguards and the proliferation resistance of current and future nuclear fuel cycle systems, and for the security of nuclear and radioactive materials. It tackles R&D challenges in the areas of nuclear materials measurement, containment and surveillance, and process monitoring. In any event, activities carried out in Europe should be coordinated to ensure effectiveness. Accordingly, it is recommended below that the EU should start an initiative to review, coordinate and elaborate on current European verification activities with a view to setting up a European Centre for Nuclear Verification Research.

The benefits of European engagement

As noted above, several initiatives related to nuclear disarmament verification are currently under way. Besides the UK–Norway Initiative, there is the 'International Partnership for Nuclear Disarmament Verification' initiated by the USA, which brings together NWS and NNWS under a cooperative framework but only includes US allies and partners.[42] While there is a need for continuing progress on an international scale, there are also concrete benefits from cooperation that engages a multitude of European states. The EU community would benefit from an accelerated development of innovative verification technologies to become a more prominent player in the multilateral dismantlement discussions.

Dismantlement expertise of European nuclear weapon states

Significant expertise linked to the dismantlement activities of European NWS already exists. These will be helpful for the verification of nuclear warhead dismantlement but also for nuclear disarmament in a broader context: since the end of the cold war, France and the UK have carried out irreversible

[37] Bird, G., Permanent Representative of Australia to the United Nations, 'Statement on the humanitarian consequences of nuclear weapons', 30 Apr. 2015, <http://www.un.org/en/conf/npt/2015/statements/pdf/HCG_en.pdf>.
[38] See Mogherini (note 1).
[39] EU/E3+3 and Iran, 'Joint Comprehensive Plan of Action', Vienna, 14 July 2015, <http://www.eeas.europa.eu/statements-eeas/docs/iran_agreement/iran_joint-comprehensive-plan-of-action_en.pdf >
[40] NPT/CONF.2010/50 (note 37), p. 24.
[41] Bylica, J., Principal Adviser and Special Envoy for Non-proliferation and Disarmament, European External Action Service, 2015 Review Conference of the Parties to the Treaty on the Non-Proliferation of Nuclear Weapons, 'EU Statement: United Nations Treaty on the Non-Proliferation of Nuclear Weapons', Main Committee I, 1 May 2015, <http://eu-un.europa.eu/articles/fr/article_16410_fr.htm>.
[42] US Department of State, 'An International Partnership for Nuclear Disarmament Verification', Fact sheet, 4 Dec. 2014, <http://www.state.gov/t/avc/rls/234680.htm>.

disarmament measures. France has dismantled its nuclear test site in Mururoa and its silo-based nuclear deterrent, and converted its fissile material production facility. Both NWS reduced their nuclear-related submarine and air components. Due to their civilian and military nuclear establishment, both countries have access to much knowledge, sophisticated equipment and proven procedures that will be instrumental to future dismantlement facilities. This expertise would for example be highly relevant for a dedicated dismantlement facility, as proposed by the British Pugwash Group in November 2012.[43] To this can be added the expertise of countries such as Sweden, which progressed quite far in developing elements of nuclear weapon facilities but abandoned the option in the 1950s and 1960s. It dismantled or converted facilities, but retained some specialist expertise on radiation protection and verification, which later flowed into international organizations such as the CTBTO.

NWS–NNWS cooperation

For multilateral disarmament to be possible, it is obvious that a future verification regime and R&D cooperation would necessarily require cooperation between NWS and NNWS. Cooperation in an European context could bring various stakeholders together to review, coordinate and enable multilateral dismantlement R&D within Europe. The UK–Norway Initiative is a successful example. Their first managed access exercises, in which the UK played the role of the NNWS and Norway the NWS, were carried out in 2008 and 2009. A warhead was simulated by using a Co-60 source. In a later exercise, carried out in 2010, the roles were reversed.[44] A general benefit of reversing the roles and developing equipment for verifying Co-60 is the reduced risk of unwillingly sharing sensitive information, such as warhead design information. Successful cooperation between a NWS and NNWS would require sufficient time for a build-up of trust and common ground. European countries in general have similar cultures and attitudes. It is more difficult to envisage a corresponding cooperation between more adversarial states with different cultural backgrounds. While such cooperation would also be helpful, the challenges of engaging more European NWS and NNWS in a similar cooperation would be less severe and would be a good continuation in the right direction.

Unique EURATOM experience

The EURATOM safeguards system, established by the EURATOM Treaty in 1957, is a set of controls and verification activities that cover all civil nuclear installations throughout the EU. Nuclear facilities in the military domain or related to national security are excluded from the EURATOM Treaty. However, in contrast to the IAEA, EURATOM has similar rights of access to all civilian nuclear facilities in both the NNWS and the NWS of the EU. In addition, EURATOM has experience of the conversion of facilities producing material for both nuclear weapons and civil purposes to exclusively civilian production facilities.

EURATOM runs a research programme on nuclear research and training under the European Framework Research Programmes. One of the objectives of the current EURATOM research and training programme (2014–18) is to 'improve nuclear security including: nuclear safeguards, non-proliferation, combating illicit trafficking and nuclear forensics'.

Existing technical expertise in Europe

To enable NNWS participation in disarmament verification, technical capacities must be built independently to create nuclear verification expertise that does not depend on the input of NWS alone. States with civil nuclear energy programmes usually have expertise in nuclear physics, nuclear engineering and radiation detection. This knowledge is highly relevant and required for the development of disarmament verification tools. It is not, however, sufficient on its own, but must be complemented by an understanding of specific disarmament verification challenges such as dealing with managed access and sensitive information. States that do not have this general level of nuclear expertise but do have an interest in participating in verification activities should also eventually be enabled to reach a level of expertise and capacity that allows for their meaningful participation. To make the required progress on the technical R&D agenda, cooperation should certainly include scientists and engineers who already have advanced knowledge.

The EU has significant expertise in the nuclear field: 17 states have had, currently have or plan to have civilian nuclear energy programmes. Research reactors either have been or are currently operated in four more states. This leaves only seven states with no expertise. At the EU level, in addition to the EURATOM safeguards inspectorate, the JRC has a wealth of expertise. The Institute for Transuranium Elements employs 370 academic, technical and support staff, and has developed an extensive range of advanced facilities over more than 50 years.[45] Its Nuclear Safeguards and Forensics, and Nuclear Security Units are of particular relevance.

IV. Conclusions and recommendations on European cooperation

Section III lists current European research activities in the fields of nuclear security, nuclear safeguards and non-proliferation. It is clear that, among them, R&D initiatives and expertise exist that are also related to potential nuclear disarmament verification technologies. Some of these initiatives, such as the related activities of the JRC, are funded by the European Commission. Others, such as ESARDA,

[43] Anderson, B. et al., *Verification of Nuclear Weapon Disarmament: Peer Review of the UK MOD Programme* (British Pugwash Group: London, Nov. 2012).

[44] The UK–Norway Initiative, 'Further research into managed access of inspectors during warhead dismantlement verification', 53rd INMM Annual Meeting, Orlando, 2012.

[45] On the work of the JRC, see <https://ec.europa.eu/jrc/en/about/itu>.

are based on European and national support; while still others are funded as part of national research grants for nuclear safeguards and non-proliferation R&D activities.

Section III also discusses activities that focus exclusively on disarmament verification technologies, carried out by, among others, Germany, Norway and the UK, partly or jointly in cooperation with non-European partners. Although these activities have been small in number and at times budget, some have had a significant impact. However, thus far there has been no overall EU strategy on R&D in support of multilateral nuclear disarmament verification.

There are multiple arguments in favour of increased and coordinated European engagement and that demonstrate why Europe seems to be destined to play a leading role in future multilateral disarmament verification, with regard to both research and policy.

1. The EU has stated its interest in verified nuclear disarmament on numerous occasions. Such statements could be meaningfully underscored by supporting related R&D activities. If the EU had a coordinated, comprehensive R&D initiative aimed at investigating and advancing disarmament verification methodologies and technologies, it would further leverage international recognition of the EU's commitment to nuclear disarmament.
2. Successful nuclear disarmament must take a multilateral approach that also includes NNWS. European states represent this required diversity. In particular, a NWS–NNWS cooperation is feasible because there is sufficient trust to enable such cooperation, especially among the EU member states.
3. The EU has significant and unique experience that is instrumental to effectively advancing the disarmament verification agenda. Specifically, this includes the dismantlement expertise of France and the UK and the expertise of EURATOM in the verification of former military facilities and materials, as well as the significant R&D efforts by the JRC in regard to nuclear safety, security and non-proliferation. Significant technical expertise also exists among the non-nuclear weapon EU member states, linked to their nuclear energy programmes.

A number of recommendations are set out below on how to increase the expertise and visibility of the EU in its support of future global nuclear disarmament, the Global-Zero process, future FMCT negotiations and other future arms control and disarmament agreements. The recommendations contain actions that mainly focus on R&D methodologies and technologies. While these touch on policy questions and could lay the basis for different decision-making pathways in disarmament policy, the technical issues are emphasized.

1. Some aspects of the current research programme of the JRC focused on nuclear safety, security and non-proliferation might also be applicable to disarmament verification. It should be evaluated, to what extent past and current activities have been relevant for disarmament verification
2. In the past, EURATOM has taken over responsibilities for safeguarding former military fissile materials in the EU member states. These activities can provide unique experience of relevance to nuclear disarmament verification. As there has never been a detailed study or process to evaluate these experiences, this should be initiated, and the lessons learnt, if appropriate, fed into disarmament verification research activities.
3. Although global disarmament policy may generally be stalled until the next NPT Review Conference in 2020, further understanding of and technical solutions to the complex challenges implied in the verification of future disarmament agreements cannot be delayed. The EU should therefore provide appropriate support for the development and application of new technologies or concepts, in addition or as a significant contribution to the International Partnership for Nuclear Disarmament Verification. In this context, the European Commission should extend the scope of appropriate funding mechanisms to disarmament verification methodologies and technologies.
4. The US-led nuclear security summit process aims to reduce the amount of existing fissile material and secure it. Disarmament and its verification can also help to reduce the amount of and secure weapon-usable nuclear material. The EU Chemical, Biological, Radiological and Nuclear Risk Mitigation Centres of Excellence Initiative could be considered a platform for intensified R&D in support of multilateral nuclear disarmament verification.
5. We propose the commencement of a one-year review process in Europe that includes all relevant national and European institutions, research groups, authorities (Euratom, IAEA) and associations (ESARDA, European Physical Society) in order to create a coordinated multilateral disarmament capacity in Europe. The results of this one-year review should be presented and discussed at an international workshop. The results could also lead to a European contribution to the NPT Action plan (Action 19), agreed at the 2010 NPT Review Conference.
6. Although the expertise gathered from the various institutions and R&D coordination through the review process would result in important progress, the establishment of an EU Centre for Disarmament Verification would maximize the effectiveness of R&D and the communication of results.

Acknowledgements

The authors would like to thank Giorgo Franceschini (PRIF, Frankfurt) for his encouragement to write this paper, all the other members of the German Disarmament Verification Network for the fruitful discussions and interesting presentations and Gerald Kirchner (ZNF, Hamburg) for his useful comments on the paper.

LC-AB-14-053-EN-N

www.esarda.eu