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Contents

Considerations for Use of Physical Protection System Components in International Safeguards Applications

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Abstract

Physical protection systems and components have potential cost and functionality advantages for use in international safeguards applications. To realize these advantages, however, the differences between physical protection and international safeguards environments must be taken into account. These differences include their differing threats, timeliness requirements, design principles, and system objectives. This paper includes a review of some physical protection system components that could be useful in international safeguards applications along with proposed guidelines for their selection and use.

1. Introduction

Whereas in the nuclear weapons states physical protection and safeguards are closely linked together under government responsibility, in the non-nuclear weapons states parties to the Treaty on the Non-Proliferation of Nuclear Weapons safeguards are implemented and executed by the International Atomic Energy Agency (IAEA). Therefore, specific requirements apply to safeguards instrumentation, which normally exceed the user requirements for physical protection instrumentation. With the emergence of the IAEA Strengthened Safeguards System under the 1997 Model Protocol Additional to the Agreement(s) between State(s) and the IAEA for the Application of Safeguards (INFCIRC/540, corrected), the whole technical control system of the IAEA is being reviewed under the aspect of safeguards effective and efficiency. This includes also an assessment of instrumentation applied elsewhere with a potential for safeguards.

In the European Union, the European Safeguards R&D Association (ESARDA) is reviewing the situation and providing advice to the EU member states. To this end the ESARDA Working Group on Containment and Surveillance developed an interest in studying the technical characteristics of physical protection instruments and devices and investigated the possibilities for transfer of technology from the physical protection to the safeguards areas. The present paper is a contribution to this investigation and reflects aspects of the discussions within the Working Group on Containment and Surveillance. Much of the following physical protection information and descriptions of physical protection components are adapted from an international training course on physical protection of nuclear facilities and materials [1].

2. Background

Using physical protection equipment in international safeguards applications could provide significant

benefits over currently available, dedicated international safeguards systems. The larger inventory and variety of physical protection components, both hardware and software, would provide greater flexibility for designers of international safeguards systems. The greater experience and testing base of physical protection equipment would provide international safeguards designers and users with greater confidence in their systems' reliability and effectiveness. The much larger market for physical protection components and systems could also result in significant cost savings in international security systems.

To achieve these advantages of using physical protection components and systems in international safeguards applications, the difference between the two applications areas must be fully understood and appropriate modifications made. The major differences between physical protection and international safeguards can be categorized by the threats they are intended to address, timeliness requirements, design principles, and system objectives.

3. Comparison of System Objectives and Design Principles

3.1. Common Design Approach

In general, while some of the terminology varies slightly, both physical protection and international safeguards use a similar approach in developing a system design. This includes the following steps:

1. Define the threat or diversion scenario
2. Characterize the facility or Material Balance Area
3. Perform the analysis that drives the technology to be employed
4. Complete the design
5. Reanalyze the design to ensure that it adequately addresses the original threat or diversion scenario

3.2. Threat

While physical protection applications address both insider and outsider threats, they are not primarily intended for applications where the host is the recognized threat. That is, in physical protection systems the operators of a facility are assumed to be interested in protecting the facility and its contents from personnel whom they do not authorize for access. Thus, some physical protection systems rely on such provisions as machine-readable badges, personal identification numbers (PINs), biometrics, etc., to identify personnel wanting access to the facility and determining whether or not they are authorized.

While international safeguards is interested in protecting materials from unauthorized access, it is also concerned about activities which from the point of view and protection schemes of the facility owner are authorized, but from the point of view of the international community would be considered inappropriate/unauthorized. The operation of a physical protection system is generally the responsibility of the facility operator. The operation of an international safeguards system is outside the facility, since the primary threat is from the operator of the facility or other authorized personnel. Therefore a traditional physical protection system may be useful in international safeguards applications to detect part of the problem – unauthorized activity (from the facility's point of view), it will do nothing (and cannot be trusted) to detect authorized activity (from the facility's point of view, but unauthorized from an International perspective). As a result, many identification and authorization technologies, such as PINS, biometrics, card readers, and proximity badges are less effective for Safeguards applications.

3.3. Timeliness Requirements

Timeliness of response is a stringent requirement on physical protection systems. The operator requires notification of an intrusion in time to allow security forces to respond in less time than it takes the adversary to accomplish its goal. This generally means that the system must provide real-time information of sufficient quality for the security force to make response decisions quickly.

The timeliness requirements on international safeguards systems are significantly different. The

international safeguards system requires the timely recording of sensor information. In essence, the monitoring system must respond in near real-time. The requirement is for the monitoring party to be notified of diversion of materials before the adversary has had time to incorporate them into a weapon. The timeliness criterion is weeks or months rather than minutes.

3.4. Design Principles

Physical protection systems designs are based on three characteristics:

1. Protection in depth. The system doesn't rely on one technology or sensor – video motion, radiation counters, motion sensors, seals – all of these are employed to provide protection in depth and layers of protection.
2. Minimum consequence of component failure, again emphasizing the need for redundancy or fail safes.
3. Balanced Protection. The design is driven by the possible ways to divert material – steal the container, take some of the contents, divert movements from normal, etc. Balanced protection implies each of these avenues has protection/monitoring capabilities.

These design characteristics provide multiple levels of detection and delay to mitigate the effects of a determined adversary's circumventing any one of them.

In international safeguards systems, redundancy, dual and containment and surveillance are employed to allow continued system operation after single-point component failures.

3.5. System Objectives

The differences in requirements and design between physical protection and international safeguards systems derive from the different objectives of their operators. Physical protection systems exist to prevent nuclear sabotage or theft of nuclear materials (ITC reference). International safeguard systems are intended to provide timely detection of diversion of *significant quantities* of nuclear materials and deter such diversions by imposing a risk of early detection.

4. Functions of Physical Protection Systems

4.1. Detection and Assessment

Detection is the discovery of abnormal activity in the protected area. Physical protection systems employ a variety of sensors for detection, including break-beams, door switches, motion detectors, infrared sensors, and video systems. Once an abnormal condition has been sensed, detection requires that the event be communicated to the system's operators. This function includes appropriate display and reporting to allow assessment and response.

In the assessment function, an operator reviews detection and other information to judge whether the event is an actual intrusion for which response is required. Automated information processing may be employed to assist in the decision-making process.

4.2. Delay

Delay refers to the ability of the physical protection system to slow the adversary's progress long enough for an effective response to be mounted.

4.3. Response

Response includes actions aimed to interrupt and neutralize the adversary's activity. These actions can be taken by human response forces or by automated means. Figure 1 below depicts the activities

and time involved for an adversary to access the target and complete their actions, in comparison to the Physical Protection System time requirements designed to preclude the adversary's success. Obviously, the Physical Protection System, to include detection, assessment, delay and response must take less time than it takes the adversaries to accomplish their goal.

As stated earlier, the timeliness requirements on international safeguards systems are significantly different. The requirement is detect the diversion of materials before the adversary has had time to incorporate them into a weapon. The timeliness criterion is weeks or months rather than minutes.

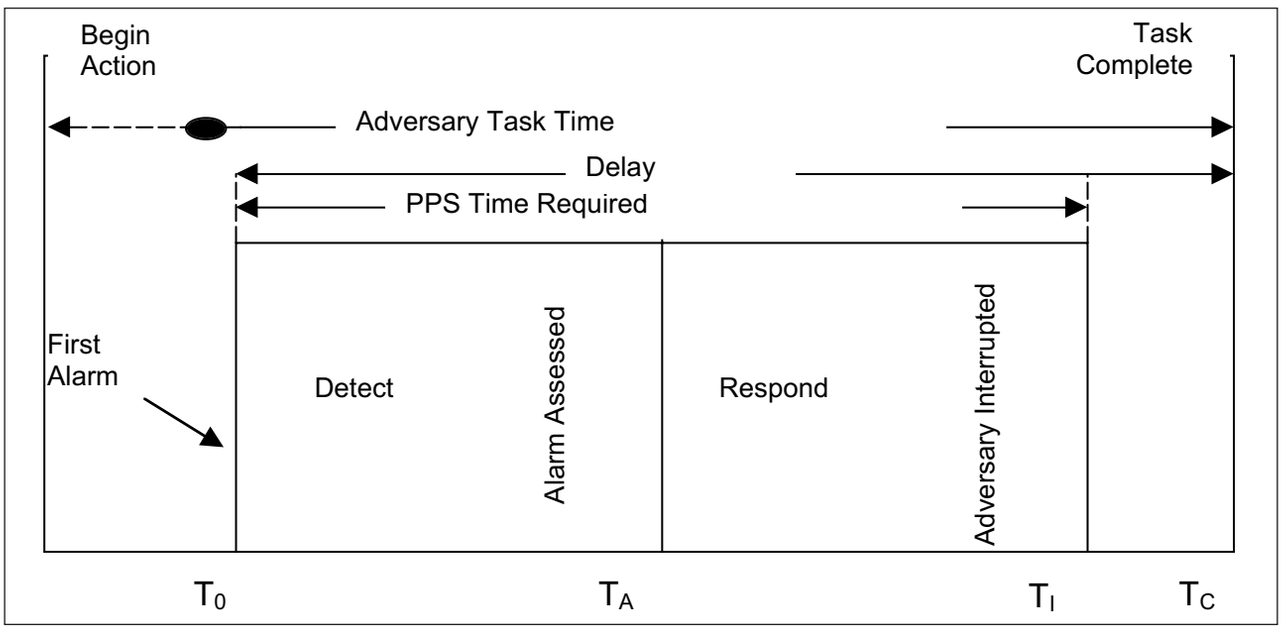

Figure 1: Adversary Task Time vs. PPS Time Requirements

4.4. Focus of this Paper

Due to the differences in timelines for system response, Physical Protection components designed to achieve delay and response will not be considered in the remainder of this paper. The primary emphasis will be on detection system components. Within the category of detection system components, due to the differences in the recognized threat for Physical Protection Systems (primarily outsider threats) versus International Safeguards (the host national entity), Physical Protection components that are inherently vulnerable to a host-threat will not be considered in the remainder of this paper. This excludes personal identification components such as PINS, biometrics, card readers, and proximity badges. While perimeter components such as fences, locks and guards may have applications for specific international safeguards applications, they too are more susceptible to a host nation threat, and will not be considered during this initial analysis. Therefore, the primary focus of the remainder of this paper will be interior intrusion detection components. Future analyses may beneficially expand into the component areas not covered by this paper.

5. Physical protection components and potential applications to International Safeguards

5.1. Overview

Physical protection components may be divided into three categories with respect to their applicability to international safeguards. The first category includes physical protection components that are worth considering for specific applications, but were not identified for potential wide safeguards

use. This may be due either to limited applicability for safeguards applications, vulnerability to a host-threat scenario, or high false alarms rates. The second category includes those physical protection components that appear to have potential for safeguards applications. They are described along with their potential applications. Further development that might be necessary for their successful use in safeguards is also discussed. The third category includes physical protection components that are already in widespread use for safeguards. These are listed, but not discussed in detail.

5.2. Physical Protection Components with Limited Potential for Safeguards Applications

Vibration Sensors. These are passive, visible, line or point sensors that detect movement of the surface to which they are attached. They detect movement of the surface to which they are attached. They range from simple jiggle switches to complex inertial or piezoelectric sensors. They can be designed to distinguish between frequencies associated with breaking and entering and normal building vibrations. Glass break sensors are in this category. They are subject to false alarms if there is significant ambient vibration in the immediate vicinity. For certain static storage applications, vibration sensors might be effective for Safeguards applications. These applications could possibly include:

- Material access
- Material movement
- Equipment tamper indication

Passive Sonic Sensors. These are passive, volumetric sensors. They are among the simplest intrusion detectors, using a microphone to detect the sounds generated in the area. If sounds of the amplitude, frequency content, and duration or repetition rate corresponding to a destructive penetration are detected, an alarm is generated. It is possible to make the sensor respond only to frequencies in the ultrasonic range. Depending on how they are attached, passive sonic sensors may be of limited effectiveness detecting only low-skill, noisy intruders. They should not be installed close to inside noise sources or near outside walls or doors where exterior noises could be a source of nuisance alarms.

Pressure Sensors. These are passive, covert, line or point detectors. Pressure sensors, usually in the form of mats, are placed around or under the protected object. They consist of a series of ribbon switches positioned parallel to each other. Ribbon switches are constructed from two strips of metal in the form of a ribbon separated by an insulating material. When an adequate amount of pressure is applied, for instance by an intruder stepping on it, the metal strips make contact and initiate an alarm. Pressure sensors should be well concealed, such as under carpets, tile or linoleum floors. If an intruder is aware of their existence, they can be easily avoided or defeated. Therefore, they are particularly susceptible to a host threat, and should only be used to detect low-skill intruders.

Light Sensors. These monitor the average light level within the field of view. If the light level changes by a predetermined amount, the possibility of an intrusion exists. The light sensor is designed to produce an alarm when such a change occurs. For certain static storage or enclosure applications, light sensors might be effective for Safeguards applications.

5.3. Physical Protection Components with Greater Potential for Safeguards Applications After Further Development

Active Infrared Sensors. These are active, visible, and line or plane sensors that establish a beam of infrared light using an infrared light source as the transmitter and photodetectors for receivers. Several transmitters and receivers are usually employed together to provide a system with multiple beams, configured vertically as an infrared fence. A pulsed, synchronous technique may be used to reduce interference and the possibility of defeat by other sources of light. The narrow vertical plane in which this sensor operates does not provide any significant volume coverage, and the system designer must carefully consider its installation in order to avoid easy defeat or bypass. Infrared sensors may be susceptible to several nuisance alarm sources, including smoke and dust in the air, which can scatter the beam to the point where an alarm is initiated. However, the amount of smoke

and dust required to scatter the beam is significant. As part of a total sensor package, active infrared sensors may prove effective.

Infrasonic Sensors. These sensors operate by sensing pressure changes in the volume in which they are installed. A potential application is to provide tamper detection inside an equipment rack or junction box. Opening or closing the door would result in a slight pressure change and generation of sound pressure waves with frequencies below 2Hz. Passive infrasonic sensors respond to such waves, but air blowing into the closed volume can cause nuisance alarms.

Specific required development for use in safeguards applications includes the following:

- Testing to determine if cooling units inside equipment racks (with the accompanying sound and air movement) would create false alarms.
- Adjusting sensitivity to detect opening of the rack, but not generate false alarms by the operation of the equipment within the rack.
- There is a concern with the vulnerability of infrasonic sensors to slow motion diversion attempts. This would require evaluation.

Capacitance Proximity Sensors. These are most commonly used as proximity sensors, but can be used as boundary penetration sensors as well. Capacitance proximity sensors are active, covert, line or plane sensors. They detect intruders approaching or touching metal items or containers that the sensors are protecting.

In a capacitance sensor installation, capacitance between a protected metal object and ground comprises part of the total capacitance of a tuned circuit in an oscillator. The electrical conductivity on an intruder's body alters the dielectric characteristic of the protected space as the intruder approaches or touches the protected object. The dielectric change results in a change in the capacitance between the protected item and the reference ground, resulting in an alarm.

The object to be protected is normally not grounded. For applications where the object to be protected must be grounded, the object can be considered to be the ground plane. This requires the fabrication of a capacitance blanket to drape over the protected object (see the figure). If the blanket is made large enough to cover the object entirely, any access attempts will cause blanket movement, capacitance change, and an alarm.

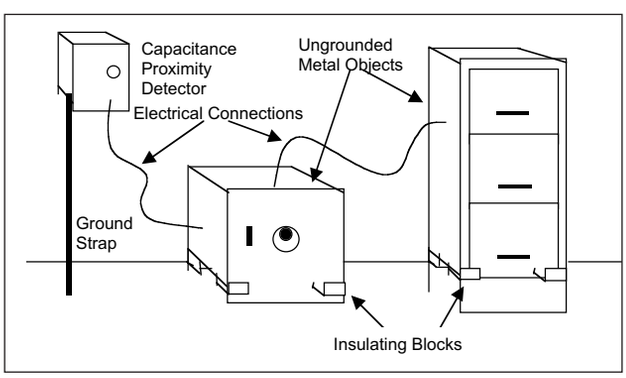

Figure 2. Typical Capacitance Proximity Detector Connections for Ungrounded Objects

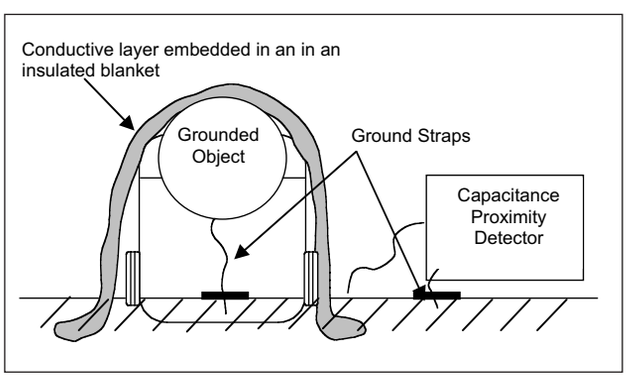

Figure 3. Grounded object protected by a capacitive blanket

Capacitance proximity sensors are used to detect boundary penetration through existing openings such as grills and ventilation ducts or metal window frames and doors. They can detect penetration of:

- Equipment racks
- Junction boxes

- Fresh fuel racks
- Sealed vaults
- etc.

Specific required development for use in safeguards applications includes the following:

- Testing to determine if sensitivity can be adjusted to distinguish between penetration of a protected container or room versus the movement of equipment or people just outside the boundary.
- Testing of means to compensate for effects of relative humidity and the proximity of other metal objects.

Electric Field Sensors. These are similar to capacitance proximity sensors except they may cover larger areas. They consist of sets of wires along a wall or surface which alarm when a human approaches or touches the wires.

Like capacitance sensors, they may be effectively used to detect penetration of:

- Fresh fuel racks
- Sealed vaults
- etc.

Because of the physical configuration of these sensors, they are less practical for smaller objects such as equipment racks and junction boxes.

Specific required development for use in safeguards applications includes testing to determine if radiation environments affect electric field sensors. Furthermore, nuisance alarms may be significant if there are fluctuations in electric fields in the vicinity – power cables, motors, etc. There is also a concern with the vulnerability of electric field sensors to slow motion diversion attempts. This would require evaluation.

Common Required Development.

Common required development for each of the above components to be used in safeguards applications includes the following:

- Authenticating sensor data. If the sensors are used inside a TIE, sensor data could be authenticated and buffered through a separate processor co-located inside the TIE. If the sensors are not installed inside a TIE, sensor data would have to be authenticated and buffered or passed through a secure conduit to a separate processor where these functions could occur.
- Conducting a vulnerability assessment to certify use for specified applications.

5.4. Physical Protection Components Currently in Widespread Use for Safeguards Applications

Some physical protection components are already in wide use in safeguards applications. These include:

- Fiber-optic cable sensors
- Balanced magnetic switches and other electromechanical sensors
- Interior motion sensors
- Video motion sensors

Due to their current widespread use in safeguards, these sensors are not further addressed in this paper. There are potential benefits to be gained, however, by applying inexpensive physical

protection components in new ways in international safeguards applications. For example, an inexpensive (approximately $50 per unit) digital IR camera could be placed inside a tamper indicating enclosure to capture images of a panel being opened. Applications such as this should be included to a follow-on analysis to this paper.

6. Summary

Physical protection systems and components have potential cost and functionality advantages for use in international safeguards applications. This paper reviewed physical protection system components that could be useful in international safeguards applications along with proposed guidelines for their selection and use, taking into consideration such factors as differing threats and timeliness requirements.

Physical protections components which demonstrated the most immediate potential for benefits in international safeguards applications included:

a. Active infrared sensors
b. Infrasonic sensors
c. Capacitance proximity sensors
d. Electric field sensors
e. Radiation sensors

It was acknowledged that other sensors could also provide benefits when used in very specific type applications. These included:

a. Vibration sensors
b. Passive sonic sensors
c. Pressure sensors
d. Light sensors

It was also acknowledged that some benefit could be gained by using inexpensive physical protection components in new ways for international safeguards – such as the use of inexpensive digital IR cameras.

Next Steps: To gain the potential benefits posed in this paper, the following steps should be taken:

a. Perform the evaluations recommended in the paper on the components that demonstrated the greatest potential for international safeguards use. Successful evaluations could yield the confidence level needed to initiate a field test or actual application.
b. Perform evaluations of selected components that demonstrated limited potential for international safeguards use. Successful evaluations could yield the confidence level needed to initiate a field test or actual application.
c. Continue to evaluate other physical protection areas, such as exterior sensors and assessment components for potential value.
d. Investigate the use of simple, inexpensive components in new ways to enhance international safeguards applications.

Reference

[1] 16th International Training Course: Protection of Nuclear Facilities and Materials. Sandia National Laboratories, Albuquerque, NM, 2002

Report on the Joint C/S and NDA Working Groups' Meeting on Unattended and Remote Monitoring and Measurement Systems

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1. Introduction

The ESARDA Working Groups (WG) on Containment and Surveillance (C/S) and on Non Destructive Assay (NDA) held a joint meeting in Brussels on 30th October 2002.

The purpose was to discuss aspects of unattended and remote monitoring systems, paying particular attention to matters of common interest and to the interface between C/S and NDA.

On the part of the C/S WG there were 10 attendants, the NDA WG was represented by 19 attendants. The European Commission's safeguards authority was represented by one person, the International Atomic Energy Agency (IAEA) by 2 persons, and the Argentine Brazilian Agency for Accounting and Control of Nuclear Materials (ABACC) by one person. US national laboratories delegated three persons.

The groups discussed the document entitled "Guidelines for Unattended and Remote Monitoring Systems (URMS)" (version of March 21, 2002), drafted by the C/S WG, with the intention to issue a joint document. Furthermore, there were presentations and discussions on: (1) the outcome of the recent IAEA Workshop on Standardisation and Integration of URMS; (2) the status of implementation of URMS and future needs; (3) authentication, encryption, authorisation, data collection, remote transmission, and review; and (4) system architecture.

2. Discussion

In the discussion the working groups addressed all relevant issues which arose in connection with the aforementioned topics. With reference to the draft URMS guidelines the opinion was expressed that the quality control aspect was not sufficiently covered in the document. This was explained by giving the following example. If a radiation detector system is used to perform a quantitative measurement rather than monitoring the presence of radiation only, there are two aspects to ensuring the proper operation of the system: (1) checking the performance of the system and (2) checking the correctness of the response. It is this latter aspect which still needs to be addressed in the URMS guidelines.

Secondly, authentication is a difficult issue in NDA, as, generally, sensor and data generator are separated from each other with a signal transmission line in between bridging the spacial distance. Present state of the art requires that the authentication takes place at the data generator level. If it

is not possible to mount both sensor and data generator in a single cabinet, the perception is that there is no solution to the problem.

The IAEA representative pointed out that the majority of unattended systems was not used to perform quantitative NDA measurements but radiation monitoring activities. However, a few quantitative systems do exist, an example being an unattended system in Japan for measuring true neutron coincidences for plutonium determination. The IAEA also uses the facility operator's isotopic composition measurement system.

It has been stated at the meeting that the trend for the future is that the number of NDA systems used as unattended and remote *measurement systems*, i.e., in "quantitative" mode, will rapidly grow.

Authentication is an important issue for the IAEA. Also the Euratom representative stressed the necessity of authentication, i.e., ensuring the authenticity of date and time annotation, content and origin of the data. The concept of an integrated radiation detector/data generator unit has been realised in Kazakhstan, in a location with a low radiation background, where all components are mounted in a single box. The problem is neutron radiation inducing hardware and software failures in the data generator. It is difficult to apply appropriate shielding. Another authentication approach has been proposed by a Canadian company. To detect a possible radiation attack the concept uses an internal pulse as a reference in connection with a motion detection sensor. In the IAEA it is believed that authentication concepts for NDA signals may differ from those for images or electronic seals.

A developer recalled that authentication was previously authorised and implemented in analogue camera systems. Cameras deployed in the present systems do have analogue sensors which, like NDA detectors, are sensitive to radiation. Also, there may be black images which is equivalent ot an incorrect response of the system and, hence, a quality control issue. Therefore, there is no real difference between camera surveillance and NDA measurements.

Furthermore, the case is addressed where the IAEA may want to share an unattended NDA system with the facility operator. The parties may share the detector signals or the data. Also here, authentication is required. Authentication must be developed for all radiation detection and monitoring systems, i.e., gamma, passive and active neutron monitoring systems.

In Germany, a digital unattended multichannel analyser (DIUM) is being developed which will meet the requirements of a data generator with authentication and encryption. In order to avoid radiation damage to the DIUM, the inspectorates have rejected the proposal to integrate the data module (i.e., the DIUM) and the sensor unit (i.e., gamma detector and preamplifier) in a single housing. The project does not foresee authentication for preamplifier signals.

According to a developer these signals could be transmitted through a mono mode fibre optic cable which would provide a high security level if installed between the preamplifier and the data generator module. The cost aspect should not be ignored. A developer stated that a cost analysis cannot be performed without knowledge of the technical solution for authentication. Another participant stated that CEA (the French Atomic Energy Commission) has developed the ADONIS system which is a MCA applied to high counting rates. Here, a digitisation module is integrated in the detector head. It is well adapted to authenticate the signal at the sensor level.

Regarding the finalisation of the URMS guidelines document, the NDA WG was asked to provide input to make it a joint document of the C/S and NDA WGs. Notice is taken that the IAEA will pursue their own document. The NDA WG will add aspects of the quantitative assay mode of NDA systems. Furthermore, it was agreed to add a chapter dedicated to definitions and a diagram of a generic system (system architecture). In the future, the document must be revised as required, in order to account for the technical evolution. The WGs anticipate that the document could be used by the Euratom authority in connection with invitations to tender.

3. Summary and Conclusions

The joint working groups discussed aspects of Unattended and Remote Monitoring Systems (URMS), paying particular attention to matters of common interest and to the "interface" between

C/S and NDA.

The groups discussed also the following document :

"Guidelines for Unattended and Remote Monitoring Systems"

drafted by the ESARDA Working Group on C/S, version of March 21, 2002, and decided to have input from the ESARDA Working Group on NDA, in order to reach an agreement on a joint document incorporating C/S and NDA views.

In the above document NDA (radiation) sensors are considered almost only as sensors in surveillance, i.e., monitoring mode. However, several unattended systems are currently operated, both by Euratom Safeguards and IAEA, in which NDA equipment is used also for quantitative assay. This trend will certainly continue in the future, when more and more NDA unattended equipment will be operated in quantitative, i.e., assay mode.

NDA in surveillance/monitoring mode is defined as using the radiation monitor response to survey operations, e.g., by triggering cameras when a prescribed level of radiation is exceeded.

NDA in quantitative/assay mode is defined as using the NDA sensor signal (e.g., gamma or neutron) to derive quantitative parameters such as the Pu or 235U mass of the sample, like it is normally done with attended NDA systems.

In the scenario, where NDA is used in both assay and monitoring modes, URMS (Unattended and Remote Monitoring Systems) should better be named: URMMS (Unattended and Remote Monitoring and Measurement Systems).

Concerning NDA in the above document it was decided to incorporate aspects of NDA in both assay and monitoring modes.

The NDA WG will provide a draft of the relevant parts of the URMMS document, which should be modified in consequence of the above decision.

It was recommended that the document should also consider solution monitoring and mass/volume measuring devices, which frequently are operated in unattended/remote mode: the needs for authentication, encryption, reliability, data security, etc, are equivalent to those of C/S and NDA. The NDA WG will make a proposal.

It was decided to complement the document with two annexes: a glossary, explaining the meaning of the technical terms used along the document, and considerations on a "model" architecture for a typical URMMS, to facilitate the reading and the comprehension of the document.

A draft of the glossary definitions will be provided by the C/S WG; a draft of the "model" architecture will be provided by the NDA WG.

The following recommendations have been addressed to the NDA laboratories which work in the URMMS area:

- The aspects of authentication for NDA signals at the detector level should be studied.
- The aspects of performances, quality control (QC) and quality assurance (QA) of NDA instruments operated in unattended mode must also be re-considered

Performance values for non destructive assay (NDA) techiques applied to safeguards: The 2002 evaluation by the Esarda NDA working group

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Abstract

The first evaluation of NDA performance values undertaken by the ESARDA Working Group for Standards and Non Destructive Assay Techniques (WGNDA) was published in 1993. Almost 10 years later the Working Group decided to review those values, to report about improvements and to issue new performance values for techniques which were not applied in the early nineties, or were at that time only emerging.
Non-Destructive Assay techniques have become more and more important in recent years, and they are used to a large extent in nuclear material accountancy and control both by operators and control authorities. As a consequence, the performance evaluation for NDA techniques is of particular relevance to safeguards authorities in optimising Safeguards operations and reducing costs. Performance values are important also for NMAC regulators, to define detection levels, limits for anomalies, goal quantities and to negotiate basic audit rules.
This paper presents the latest evaluation of ESARDA Performance Values (EPVs) for the most common NDA techniques currently used for the assay of nuclear materials for Safeguards purposes.

The main topics covered by the document are:

- *techniques for plutonium bearing materials: PuO2 and MOX;*
- *techniques for U-bearing materials;*
- *techniques for U and Pu in liquid form;*
- *techniques for spent fuel assay.*

This issue of the performance values is the result of specific international round robin exercises, field measurements and ad hoc experiments, evaluated and discussed in the ESARDA NDA Working Group.

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GLOSSARY

ABACC	Brasilian- Argentinian Agency for Accounting and Control of Nuclear Materials
AGR	Advanced Gas Reactor
ANCC	Active Neutron Coincidence Counter
AWCC	Active Well Coincidence Counter
BWR	Boiling Water Reactor
BU	Burnup
COMPUCEA	COMbined Product Uranium Concentration and Enrichment Assay
CT	Cooling Time
CZT	CdZnTe
ENMC	Epithermal Neutron Multiplicity Counter
ESARDA	European SAfeguards Research & Development Association
FBR	Fast Breeder Reactor
GBAS	Glove Box Assay System
GXW	Gamma-X and Weighing
HEU	High Enriched Uranium
HKED	Hybrid K-Edge/K-XRF Densitometer
HLNCC	High Level Neutron Coincidence Counter
HPGe	High Purity Germanium
HRGS	High Resolution Gamma Spectrometry
INVS	INVentory Sample Coincidence Counter
KED	K-Edge Densitometer
LEU	Low Enriched Uranium
LWR	Light Water Reactor
MOX	Mixed Oxide
MTR	Material Test Reactor
NCC	Neutron Coincidence Collar
NDA	Non Destructive Assay
NMAC	Nuclear Material Accountancy & Control
PDGA	Passive Differential Gamma Absorptiometry
PHONID	PHOto-Neutron Interrogation Device
PNCC	Passive Neutron Coincidence Counter
PNMC	Passive Neutron Multiplicity Counter
PSMC	Plutonium Scrap Multiplicity Counter
PWR	Pressurised Water Reactor
PXRF	Passive X-Ray Fluorescence
RTGS	Room Temperature Gamma Spectrometry
SNM	Special Nuclear Material
TIMS	Thermal Ionisation Mass Spectrometry
UNCL	Uranium Neutron Coincidence Collar
UFBC	Universal Fast Breeder Counter
XRF	X-Ray Fluorescence

1. Introduction

The quantitative verification of the accountancy of fissile nuclear materials through independent measurements still represents one of the key elements of international nuclear materials Safeguards. The definition of internationally agreed performance values for the pertinent measurement techniques is an important prerequisite in this context, since international nuclear Safeguards are essentially based on the mutual understanding and agreement between the parties involved that the underlying system of measurements has well defined, documented and controllable performances, frequently referred to as: *"..latest international standards"* [1,2].

The assessment of the performances of non destructive assay (NDA) techniques represents one of the key areas where the Working Group for Standards and Non Destructive Assay Techniques (WGNDA) of the European Safeguards Research and Development Association (ESARDA) is permanently and intensively active. The wide international composition of the Working Group assures a high degree of international consensus for the evaluated performance values. Indeed, besides the ESARDA partners several observers from US, ABACC, Ukraine, Hungary are regularly participating to the various activities of the WGNDA [3]. In addition, national, regional and international control authorities are also represented in the WG, together with plant operators and R&D laboratories.

Broadly speaking, two kinds of activities have been carried out by the WGNDA concerning the evaluation of performance values: First, the most important NDA techniques have been carefully reviewed with the aim of evaluating their "typical" uncertainty, when applied to different material types in different measurement conditions. This review was supported by international round robin exercises initiated by the WG with the aim to obtain a real picture of the performances and capabilities for selected methods and techniques [4,5]. This first kind of activity led to the compilation of ESARDA NDA Performance Values (EPVs) [6].

The second area of activities related to the assessment of performances concerns common projects specifically designed and launched by the WGNDA to determine the capabilities of specific tools pertinent to NDA measurements. The latest projects in this area involved:

- Monte Carlo round robin exercises which had the aim to assess and improve modelling capabilities [7, 8]
- the ^{242}Pu Project concerning the assessment and improvement of the ^{242}Pu abundance evaluation, which led to the establishment and use of a new correlation algorithm [9,10]
- a Project conducted together with the ESARDA Low Enriched Uranium (LEU) Working Group aiming at an assessment of the sampling error for NDA measurements [11].

The continuing research and development work for NDA techniques, together with technological advances, have improved the measurement situation and the performance in a number of NDA applications during the last decade. The WGNDA therefore felt it appropriate to review his former evaluation of ESARDA NDA performance values, published in 1993, in order to provide an updated account of the progress made in NDA measurement technology. The present report summarises the most important improvements and provides, if and where necessary, updated performance values.

1.1 Definition, Scope and Use of NDA Performance Values

NDA techniques have become an indispensable tool in Safeguards, and they are presently being used to a large extent in nuclear material accountancy and control both by operators and control authorities. They are of interest to several other potential users as outlined below, and they may find even wider application within the scope of future Safeguards, where non-destructive "confirmative"-type of measurements will play an important role.

The evaluation of measurement performances for NDA techniques is therefore of particular relevance to all Safeguards "actors": not having the best information about the performances of NDA

measurement methods means losing the potential for optimising Safeguards operations and reducing costs.

The NDA Performance Values presented here are conceived to represent the "**knowledge of the overall uncertainty and sources of uncertainties associated with NDA measurement systems**", and they are intended to be used for the following purposes:

1. for the planning of inspections by Safeguards Authorities;
2. to provide guidance to users in judging the quality of their NDA measurements;
3. to decide if, under fixed settings, repeated measurements, repeated sampling or repeated inspections are an appropriate tool to reduce the overall uncertainty;
4. to analyse operator-inspector differences in Safeguards verification and accountancy;
5. to allow R&D laboratories to compare their achieved measurement performances with the latest internationally agreed PVs;
6. to define the required accuracy level of NDA standards;
7. to serve as input in accountancy error propagation models for MBAs or entire plants
8. to compare the updated performance values with earlier editions, in order to get an impression of improvements in measurement quality (change to lower values) or improvements in realistic perception (change to higher or lower values);
9. to provide orientation for national or international regulators for the definition of basic parameters (detection limits, goal quantities, anomaly definition).

The WGNDA has been fully aware of the general problems faced in the definition of NDA performance values in view of the diversity of measurement situations encountered in many NDA applications, ranging for the same determination from measurements performed with dedicated and specifically tailored equipment under controlled laboratory conditions, to measurements carried out with lower-grade portable equipment under adverse field conditions. Given this situation, it often appears difficult to quantify NDA performances just in terms of a single figure.

For this reason a range of performance values is quoted in many instances in order to take into account varying performances with measurement conditions and/or properties of the measurement items. The measurement conditions such as type of equipment, detector, counting times, etc. were also specified as far as possible in the tables providing the performance values. The values are normally rounded to one decimal digit.

It is important to note that the performance values published in this document represent measurement performances that are realistically achievable under fairly controlled measurement conditions. Where a technique is used under less favorable conditions, the actual uncertainty may be higher than published herein. On the other hand, under optimised conditions also somewhat better measurement performances than quoted in this paper might be realised in certain applications.

Furthermore, the given performance values should certainly not be used as replacement of a proper uncertainty statement for measurements performed: they should be rather considered as a kind of reference and guideline for the users in assessing the performance and quality of their own measurements.

NDA performance values for the following techniques and material types are presented and discussed in this paper:

- gamma-spectrometry, passive neutron techniques and calorimetry for the plutonium assay in unirradiated solid plutonium bearing materials;
- gamma-spectrometry and active neutron techniques for the fissile uranium assay in unirradiated solid uranium bearing materials;
- K-edge densitometry, X-ray fluorescence and gamma spectrometry for the uranium and plutonium assay in solutions;
- gamma-spectrometry and neutron techniques for spent fuels.

Salient features of the individual measurement techniques and important factors and parameters influencing their measurement performance are briefly outlined in each case. If existing, recent pro-

gress in measurement technologies and methodologies is also described. It is hoped that this additional background information will be helpful for the user in better understanding the measurement situations encountered in NDA applications and in assessing the quoted performance values.

NDA performance values for waste measurements, also included in the previous issue of the ESARDA NDA performance values, have been exempted from the present document. The Working Group felt it appropriate to address this special area of NDA applications, which has also a very large relevance for nuclear waste management, in a separate evaluation.

1.2 Origin and Structure of the Data

The ESARDA NDA performance values reported in this paper have been evaluated from data and information of different origin:

- they come from tailored laboratory experiments;
- they come from field inspection activities, re-elaborated and re-analysed with suitable statistical tools;
- they are the results of ESARDA intercomparison exercises carried out with specially designed reference materials with the aim of assessing NDA performances [4,5,12,13].

Case by case the origin of the source information is quoted in the text or referred to in the literature.

As to the expression of uncertainty, the Working Group has decided, after intense discussions, to adhere to the previous categorisation in terms of random and systematic uncertainties. This decision takes into account the fact that the majority of statistical evaluations made on Safeguards measurement data (and the conclusions drawn from them) relies on this type of classification, and that specifically for the counting techniques used in NDA the random uncertainty due to pure counting statistics often represents the dominating uncertainty component.

The adherence to the "Guide to the Expression of Uncertainty in measurement" [14] is presently becoming a standard in scientific publications. The recommended categorisation of uncertainties therein (A/B) does, however, not exclude a further use of the random/systematic categorisation. The Guide explicitly states (E3.7): "*classifying the methods rather than the components does not preclude gathering the individual components … into specific groups for a particular purpose in a given measurement, for example, when comparing the experimentally observed and theoretically predicted variability of a complex measurement system*". This is exactly one of the main purposes the present data will be used for.

The different ways of categorisation of uncertainties are therefore not at all contradictory, but rather "orthogonal" [H. Aigner, IAEA, Personal Communication], where, for the present purpose, the differentiation between random and systematic uncertainties appears to be more practical and better suited for the end users. In addition, in both systems the uncertainty propagation is performed in exactly the same way, hence, the same total uncertainty will result at the end.

The adopted definitions of the uncertainty components are also coherent with those given in: ISO, "International Vocabulary of Basic and General terms in Metrology, First Edition 1984" [15]. The categorisation of uncertainty components is also coherent with the ISO Guide 5725 [16] on Accuracy of Measurement Methods where the concept of "trueness" in statistical measurement data evaluation, has been further elaborated.

The random uncertainty component (r), which affects in an unpredictable manner the measurement result for a single item is for many NDA measurements determined to a large extent by the counting time (or of the number of repetitions). The figures given in this paper for "r" are relative to measurement times which are, generally speaking, "field" measurement times.

The systematic uncertainty component (s) contains uncertainties, which are shared by many samples, such as normalisation and calibration errors, uncertainty in nuclear constants, but also uncertainties connected with sampling and with differences in physical and chemical properties between measurement and calibration items, appearing as unknown biases.

Total (overall) uncertainty values are not given since they are generally assumed to correspond to the quadratic sum of "r" and "s" , assuming that the two types of uncertainty are independent.

All values given in the tables are relative uncertainties, given with an expansion factor k=1.

2. TECHNIQUES FOR PLUTONIUM-BEARING MATERIAL: PUO_2 AND MOX

This chapter discusses characteristic features and performances of NDA techniques utilised in Safeguards for the determination of the amount and of the isotopic composition of plutonium in unirradiated Pu-bearing solid materials. The most common NDA techniques applied for the determination of the total amount of plutonium in a measurement item are Passive Neutron Coincidence Counting (PNCC) and calorimetry. Both methods make use of isotope-specific nuclear properties and therefore do require the knowledge of the Pu isotopic composition for the interpretation of their responses in terms of the total amount of plutonium. The NDA choice for the complementary isotopic measurement is High Resolution Gamma Spectrometry (HRGS). However, very often also isotopic data from other sources such as mass spectrometry, if available, are used for the evaluation of the PNCC and calorimetry measurements.

2.1 High-Resolution Gamma Spectrometry (HRGS) for Pu isotopic composition

2.1.1 Fundamentals

HRGS represents since long time an important tool for Pu isotope abundance measurements in Safeguards. The technique, which bases on ratio measurements of isotope-specific gamma rays emitted in the natural decay of the Pu isotopes, offers the great practical advantage of not requiring a calibration with physical standards. The underlying intrinsic calibration approach relies on fundamental nuclear data such as isotope half-lives and gamma emission probabilities.

Plutonium isotope abundance measurements by HRGS can be made on virtually all kinds of plutonium samples containing unirradiated plutonium, with a wide range of plutonium masses (from mg to kg size samples). The method only requires that the sample containment offers sufficient transparency for the gamma rays to be detected for analysis. This requirement is fortunately met by a large number of sample containers for plutonium materials found in practice.

A major advancement for the measurement technique was achieved with the development of the Multi-Group Analysis (MGA) code, which successfully exploits the complex XKα region (94-104 keV) of a plutonium gamma spectrum for the isotope analysis [17,18]. Since this spectral region contains the most abundant plutonium gamma and X-rays detectable in a gamma spectrum from plutonium in the presence of some Am, fairly precise isotope abundance determinations became feasible with this analysis from gamma spectra accumulated in relatively short counting times (15-30 min).

During the last decade improved complementary analysis codes making use of the more energetic but less abundant gamma rays in the energy region between 125 and 660 keV have also become commercially available. A widely used code supporting the analysis of this energy range is the FRAM code [19]. The resort to the higher energy gamma rays for isotopic analysis provides some advantages in cases of stronger sample shielding [20]. The FRAM code has been recently upgraded to provide, like the MGA code, also analysis capabilities for the XKα region [21].

A drawback of the gamma-spectrometric technique is the lacking measurement capability for the isotope ^{242}Pu, which does not manifest itself with a detectable gamma-ray signature in a plutonium gamma spectrum because of its very low specific gamma activity. Therefore recourse has to be made to isotope correlation techniques for an estimate of the abundance of this isotope. A new type of isotope correlation relating the Pu isotope ratio 242/239 to the measurable ratios 238/239 and 240/239 has been proposed and validated from a large set of LWR isotopic data [9,10]. The new correlation provides a significant improvement for the ^{242}Pu estimate compared to previously used algorithms, especially for PWR plutonium. Nonetheless, the remaining uncertainties for the estimated ^{242}Pu abundance do still affect the overall accuracy of a complete gamma-spectrometric Pu isotopic analysis made on plutonium materials containing a notable fraction of this isotope.

2.1.2 Instrumentation

High-quality gamma spectra taken with good energy resolution are mandatory for a reliable and accurate gamma-spectrometric Pu isotopic analysis. High-resolution gamma spectrometers equipped with suitably sized HPGe detectors therefore still represent the prime choice. For the appropriate detector size, compromises have often to be made between desired optimum energy resolution and detector efficiency, but a variety of suitable HPGe are readily available.

The last decade has seen a remarkable degree of miniaturisation for the electronic part of the gamma spectrometer, resulting in light and compact portable equipment for in-field measurements. This also holds to some extent for the portable HPGe detectors, but the need for detector cooling to the temperature of liquid nitrogen during operation remains a practical obstacle for many in-field applications. Continuous efforts are therefore being undertaken with the aim to replace the cooling medium of liquid nitrogen by light and reliable electrical cooling generators.

Continuous research and development work is also going on for improved semiconductor detectors operating at room temperature or with Peltier cooling such as CdTe or CdZnTe detectors, and for the development of adequate spectrum analysis codes applicable to gamma spectra taken with such detectors. Although substantial progress and improvements have been reached in recent years, it appears unlikely that this type of detector, because of the inferior energy resolution and poorer energy response function, will become a full substitute for HPGe-based spectroscopy systems in the foreseeable future. Nonetheless, the feasibility of Pu isotopic analysis with a performance sufficient for the coarse categorisation of the isotopic grade of plutonium has been already demonstrated for measurements made with CdTe detectors [22].

2.1.3 Performance values

The evaluation of measurement performances for Pu isotope abundance measurements by HRGS has been a continuous working item for the WGNDA. To this end the WG has organised in the course of the years three inter-comparison exercises with international participation [23,24,5]. The evaluated performance values given in Table I are based on results from the last inter-comparison exercise [5], from dedicated performance studies [25,26,27,28], from information and experiences collected during a dedicated workshop [29], and from a larger number of routine laboratory measurements made for Safeguards.

The random uncertainties for the isotope abundances quoted in Table I are largely determined by counting statistics, which in turn depends on a number of measurement variables such as counting time, detector efficiency, amount of sample material, isotope ratio, type of sample shielding etc. The given performance data for the random uncertainty component refer to measurements made on samples with >1 g of plutonium in a non-strongly absorbing containment for a counting time of about 10-20 min, and for the evaluation of the isotope abundances from the XKα region. For applications involving smaller amounts of plutonium (<1 g) or strongly absorbing container materials, significantly longer counting times are usually needed in order to reach random uncertainties comparable to the systematic uncertainty components.

Systematic uncertainties are mainly associated with deficiencies in the algorithms used for the analysis of the complex gamma spectra for extraction of the required raw data like net peak areas and relative detection efficiency. The respective systematic uncertainties are difficult to quantify in general because they are dependent to some extent on the quality of the recorded gamma spectra in terms of energy resolution and quality of the peak shape. Another source of systematic uncertainty may be associated with uncertainties of the nuclear data required for the conversion of measured peak area ratios into atom ratios. The systematic uncertainties quoted in Table I represent the range of average differences between measured and declared isotope abundances as observed in a number of inter-comparisons. The data refer to HRGS measurements evaluated with declared ^{242}Pu abundances. The systematic uncertainties quoted for ^{238}Pu and ^{241}Am represent estimates, because real performance data are difficult to establish in view of the lack of accurate reference values for both isotopes.

Table I : Performance values for Pu isotope assay in PuO2 and MOX.

Type of plutonium	Isotope	r (%) [a]	s (%) [b]
Low burnup	^{238}Pu	3	5
	^{239}Pu	0.2	0.1-0.2
	^{240}Pu	1	0.3-1
	^{241}Pu	1	0.2-0.6
	^{241}Am	1	0.5
High burnup	^{238}Pu	1	1
	^{239}Pu	0.5	0.2-0.4
	^{240}Pu	1	0.5-1
	^{241}Pu	1	0.5-1
	^{241}Am	1	1

[a] Counting time 10-20 min

[b] HRGS isotopic data evaluated with declared ^{242}Pu

Table II gives typical performance values for the ^{242}Pu estimate from the new type of correlation. The separation into random and systematic uncertainty components does not apply in this case. Instead, the performance is given in terms of the typical relative standard deviation (rsd) observed for the differences between estimated and declared ^{242}Pu abundances. Performance values are quoted for 4 different types of plutonium (Magnox, AGR, BWR, PWR), to which the correlation has been applied using specific sets of coefficients for the correlation. In many instances parametric plots of the measured ratios ^{238}Pu/^{239}Pu versus ^{240}Pu/^{239}Pu allow to identify unambiguously the respective type of plutonium [10]. If the given isotopic composition does not clearly point to a particular type of plutonium (e. g, in case of mixtures), larger uncertainties have to be expected and taken into account. The observed performance for the correlation proves that it is working best for PWR plutonium.

Table II : Performance of ^{242}Pu estimate from isotope correlation.

Type of plutonium	rsd (%)
PWR	3
BWR	6
AGR	4
Magnox	4

One of the main objectives of plutonium isotope abundance measurements made by gamma spectrometry is the determination of the quantities P_{eff} and $^{240}Pu_{eff}$ as defined in Sections 2.2 and 2.3, which are required for the interpretation of the plutonium assay by calorimetry and neutron coincidence counting, respectively. The performance values given in Tables I and II have been propagated as independent variables into the respective expressions for P_{eff} and $^{240}Pu_{eff}$ to calculate the expected random and systematic uncertainties for both quantities as listed in Table III. Two sets of HRGS performance data are listed in the Table, differentiating between the origin of the ^{242}Pu

abundance: (i) declared ^{242}Pu from a TIMS measurement, and (ii) ^{242}Pu abundance estimated from the isotope correlation. The right-hand column of Table III lists, for comparison, random and systematic performance values for P_{eff} and $^{240}Pu_{eff}$ resulting from the propagation of uncertainties for Pu isotope measurements by Thermal Ionization Mass Spectrometry (TIMS)/Alpha Spectrometry according to recent International Target Values (ITV) [30].

Since the assumption of independence made for the uncertainty contributions of the individual isotopes is not strictly correct, the calculated HRGS performance values for P_{eff} and $^{240}Pu_{eff}$ may be overestimated to some extent according to practical experience. This holds particularly for the systematic uncertainty component, where the correlated uncertainties of the individual isotopes tend to partly cancel each other. For reactor-grade plutonium the performance values for P_{eff} are largely determined by the uncertainties for the ^{238}Pu abundance. This is also true for the TIMS/α data, where the current target values for ^{238}Pu abundance measurements in higher burnup plutonium (1.5 % random and 1 % systematic) would have to be tightened significantly in order to reduce their impact on the uncertainty of P_{eff}. The lack of accurate reference values for ^{238}Pu is currently also hampering the evaluation of the true performance of ^{238}Pu abundance measurements by HRGS, and hence also for P_{eff}.

Table III: Performance values for $^{240}Pu_{eff}$ and P_{eff}.

Quantity	Type of plutonium	HRGS (^{242}Pu from TIMS)		HRGS (^{242}Pu from correl.)		TIMS/α (ITV for Pu isotopes)	
		r (%)	s (%)	r (%)	s (%)	r (%)	s (%)
$^{240}Pu_{eff}$	Low burnup	1	0.3-1	1	0.3-1	0.15	0.1
	Magnox	0.9	0.5-0.9	0.9	0.5-1	0.1	0.06
	AGR	0.8	0.4-0.8	1	0.6-1	0.1	0.1
	BWR	0.8	0.4-0.8	1.3	1.1-1.3	0.15	0.2
	PWR	0.7	0.4-0.8	1	0.8-1	0.2	0.2
P_{eff}	Low burnup	0.25	0.25	0.25	0.15-0.3	0.25	0.25
	Magnox	0.5	0.3-0.5	0.5	0.3-0.5	0.4	0.3
	AGR	0.6	0.5	0.6	0.6	0.7	0.5
	BWR	0.6	0.6	0.6	0.7	0.9	0.6
	PWR	0.7	0.7	0.7	0.8	1.1	0.7

2.2 Calorimetry for Pu Mass Determination

2.2.1 Fundamentals

Calorimetry determines the amount of plutonium in an item through the measurement of the heat produced by the radioactive decay of the plutonium isotopes. Since the thermal power generated per unit amount of plutonium depends on the isotopic composition of plutonium, calorimetry requires knowledge of the relative plutonium isotope abundances for the conversion of the measured thermal power into the corresponding amount of plutonium. The calorimetric plutonium assay also needs information on the content of ^{241}Am in the measurement item, which also contributes to the measured thermal power and which as a decay product of ^{241}Pu is present in practically all plutonium samples.

The specific thermal power P_{eff} (W/g) of plutonium calculates from the expression where:

$$P_{eff} = \sum_i R_i \cdot P_i$$

R_i = abundance of the i-th isotope (i = ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{241}Am)of the Pu expressed as a weight fraction, g isotope/g-Pu, and:

P_i = a physical constant, the specific thermal power of the I-th isotope in W/g.

Table IV lists the thermal power values and their standard deviations for the plutonium isotopes, ^{241}Am and ^{3}H. The by largest specific heat is produced by ^{238}Pu. The uncertainties of the thermal power values are well below 0.1% for the main Pu isotopes. Table IV also indicates the main decay mode responsible for the heat production, which for all isotopes listed has a branching ratio of > 99.99%. The last column specifies the potential energy loss that might be lost from the calorimeter due to the escape of decay neutrons and photons. This energy loss is negligible for all isotopes except for ^{241}Am, where it could reach a maximum value of about 0.5% assuming the escape of all 59.54 keV decay photons.

Table IV: Specific thermal power values.

Isotope	Main Decay Mode	Specific Power (mW/g) [a]	Stand. Dev. (%)	Max. Energy Loss (%)[b]
^{238}Pu	α	567.57	0.05	3.1E-02
^{239}Pu	α	1.9288	0.02	1.3E-03
^{240}Pu	α	7.0824	0.03	5.5E-04
^{241}Pu	β	3.412	0.06	2.5E-02
^{242}Pu	α	0.1159	0.22	3.2E-02
^{241}Am	α	114.2	0.37	5.1E-01
^{3}H	β	324	0.14	-

[a] from Ref. [31]

[b] from Ref. [32]

The characteristic thermal power generated by different types of plutonium are listed in Table V (assuming Pu freshly separated from ^{241}Am). The Table also indicates the isotope contributing the largest fraction to the respective thermal power. For aged high-burnup PWR plutonium typically containing between about 1-3% weight % of ^{238}Pu and ^{241}Am, more than 90% of the thermal power will be generated by these two minor isotopes.

Table V: Thermal power of different types of plutonium.

Type of plutonium	Thermal power (mW/g)	Main heat contributor	Contribution in %
Weapons Grade	2-3	^{239}Pu	70-80
Magnox	4-5	^{240}Pu	35-40
AGR	5-8	^{238}Pu	40-50
BWR	6-10	^{238}Pu	50-60
PWR	10-20	^{238}Pu	60-90

The thermal power W measured from a plutonium item in a calorimeter is converted into the amount of plutonium as following:

$$m_{Pu} = \frac{W}{P_{eff}}$$

The thermal power measurement in the calorimeter is usually calibrated against electrical standards or certified plutonium reference materials (mostly ^{238}Pu heat standards). An important feature of practical relevance is the fact that the calibration samples must not be physical standards representative of the materials being assayed.

Calorimetry is a truly nondestructive method and can be applied to all kinds of plutonium bearing materials, because the heat measurement is completely independent of material and matrix type. Self attenuation does not occur. Interferences or measurement biases could be only introduced by undetected heat-generating radionuclides, or by chemical processes such as phase changes or endothermic or exothermic chemical reactions (normally not an issue for Safeguards samples). Packaging conditions of the measurement item cannot change the heat output of the sample but are usually the determining factor for measurement time. Typical assay times can range from 1-2 h up to 20 h or more. Statistical prediction algorithms are usually applied to predict thermal equilibrium and reduce measurement time.

Calorimetry is also routinely used for the assay of tritium in many physical forms of tritium compounds. For the tritium assay the measured thermal power can be directly converted into tritium mass using the specific power P_{eff} = 0.324 W/g of tritium.

2.2.2 Instrumentation

All calorimeters in use for plutonium assay are heat-flow type calorimeters, where the sample heat generated in a thermally insulated sample chamber flows past a temperature-sensing element, through a well-defined thermal resistance, to a constant-temperature heat sink. A variety of heat-flow calorimeters, differentiated by the temperature control techniques, the heat flow paths and the type of temperature heat sink, are in practical use for nuclear material measurements. The most common ones are known as isothermal “air bath”, water bath and rod calorimeter [31].

The majority of the existing calorimeters employ traditional Ni sense windings and Wheatstone bridge circuitry as temperature sensing element for measuring the heat flow. In recent years high-sensitivity thermopile sensors have become a viable alternative to the nickel thermocouple/Wheatstone bridge sensor for heat flow measurements. The intrinsically low noise and good baseline stability of the thermopile sensors make them particularly interesting for heat measurements in the milliwatt and sub-milliwatt range. Small sample calorimeters based on thermopile technology have been recently developed [33,34]. Another major advancement in terms of sensitivity and precision for low thermal power measurements has been achieved with a new generation of calorimeters, which are based on ‘‘inertial temperature control” using a metal block of high thermal inertia surrounded by heat flow sensing thermopile arrays. This variant offers an improvement of temperature control by up to three orders of magnitude, permitting the extension of calorimetric measurements down into the μW measurement range [35].

2.2.3 Performance values

The performance of a calorimetric plutonium assay depends on two basic variables: (i) the thermal power W as determined by the calorimeter, and (ii) the quantity P_{eff} as derived from an external isotope abundance measurement. The performance of the thermal power measurement is dependent on the sensitivity of the calorimeter, the baseline stability, and the item power.

Table VI lists typical performance data for the thermal power measurement obtained with taditional large sample calorimeters (air bath or water bath) equipped with Ni thermocouples and Wheatstone bridge circuitry, and with the new generation of small sample calorimeters (air bath or water bath) using thermopile sensors or combinations of thermopiles and Ni thermocouples (Hybrid calorimeters). The dominant contributions to the random and systematic uncertainties for the small sample calorimeters are due to heat distribution errors (spacial nonuniformity of the calorimeter response) and baseline fluctuations.

The total random and systematic uncertainty of a calorimetric plutonium assay is obtained from a combination of the respective uncertainty components for the thermal power and P_{eff} determination. This yields the relative combined standard uncertainty for the Pu element assay from a com-

Table VI: Performance of thermal power measurement.

Calorimeter	Thermal power level (W)	r (%)	s (%)
Large sample	0.1	0.4-0.7	0.1-0.2
calorimeter	1	0.1-0.3	0.05-0.2
(Ni thermocouple)	10	0.05-0.07	0.05-0.2
	100	0.05-0.07	0.05-0.2
Small sample	0.001	0.8-1.0	0.2-0.5
calorimeter	0.01	0.1-0.3	0.1-0.2
(Thermopile)	0.1	< 0.1	0.1

bined calorimetry and isotopic measurement in accordance with the ISO definition for total uncertainty. For reactor-grade plutonium the uncertainties for P_{eff} as listed in Table III of Section 2.1 outweigh by far the typical uncertainties from the thermal power measurement, irrespective of the method used for the isotope abundance measurement. The question of accuracy achievable for a calorimetric plutonium assay of reactor-grade plutonium simply reduces to the question of how accurate the ^{238}Pu (and ^{241}Am) abundance can be determined. For low burnup plutonium, however, where P_{eff} can be determined with much higher accuracy, calorimetry still represents the most accurate NDA technique for bulk plutonium samples.

2.3 Passive Neutron Coincidence and Multiplicity Counting Techniques for Pu Mass Determination

2.3.1 Fundamentals

The measurement of plutonium by passive neutron coincidence counting makes use of the fact that plutonium isotopes with even mass number (238, 240, 242) show a relatively high neutron emission rate from spontaneous fission. Table VII lists for the different plutonium isotopes the specific yields of fission neutrons, and the average number of neutrons, $<\nu>$, released per spontaneous fission. The actual number of prompt neutrons, ν, emitted from an individual fission event is described by the probability distribution $P(\nu)$ for the neutron multiplicity (Table VIII). This data show that in 70-80 % of the fission events two or more neutrons are released. These neutrons are emitted simultaneously and are therefore correlated in time. Delayed neutrons are neglected, since they do not contribute to the coincidence rate.

Table VII: Spontaneous fission neutron yields [41].

Isotope	Spontaneous fission yield (neutrons /s-g)	Neutron multiplicity <n>
^{238}Pu	$2.59\ 10^{3}$	2.21
^{239}Pu	$2.18\ 10^{-2}$	2.16
^{240}Pu	$1.02\ 10^{3}$	2.16
^{241}Pu	$5.\ 10^{-2}$	2.25
^{242}Pu	$1.72\ 10^{3}$	2.15

Table VIII: Neutron multiplicity distribution [41].

Probability	^{238}Pu (spontaneous fission)	^{239}Pu (thermal induced fission)	^{240}Pu (spontaneous fission)	^{242}Pu (spontaneous fission)
P(0)	0.054	0.011	0.066	0.068
P(1)	0.205	0.101	0.232	0.230
P(2)	0.380	0.275	0.329	0.334
P(3)	0.225	0.324	0.251	0.247
P(4)	0.108	0.199	0.102	0.099
P(5)	0.028	0.083	0.018	0.018
P(6)		0.008	0.002	0.003

The detection of bursts of time-correlated neutrons, which uniquely identify a spontaneous fission event among other neutron sources emitting neutrons randomly distributed in time such as (α,n) neutrons, allows to quantify the amount of plutonium in a sample. In reactor-grade plutonium the isotope ^{240}Pu usually dominates the overall emission of spontaneous fission neutrons. The primary quantity, that is commonly determined in passive neutron coincidence counting, is therefore an effective amount of ^{240}Pu, $m_{240_{eff}}$, representing a weighted sum of the amount of the 3 even isotopes 238, 240 and 242:

$$m_{240eff} = \gamma_{238} \cdot m_{238} + m_{240} + \gamma_{242} \cdot m_{242}.$$

The coefficients γ_{238} and γ_{242} describe the contributions of ^{238}Pu and ^{242}Pu to the neutron coincidence response in terms of an equivalent amount of ^{240}Pu. For the conversion of m_{240eff} into the total amount of plutonium, m_{Pu}, the weight fractions R_{238}, R_{240} and R_{242} of the plutonium isotopes 238, 240 and 242 must be known to calculate the isotope-specific quantity

$$^{240}Pu_{eff} = \gamma_{238} \cdot R_{238} + R_{240} + \gamma_{242} \cdot R_{242}.$$

With this isotopic input data the total amount of Pu then calculates to:

$$m_{Pu} = \frac{m_{240_{eff}}}{^{240}Pu_{eff}}$$

A major problem associated with many neutron coincidence measurements arises from the fact that the measured coincidence rate of spontaneous fission neutrons is not uniquely and invariably linked to the amount of the 3 even plutonium isotopes alone. Rather, in practice it is always altered (increased) by induced fission events. The magnitude of this perturbation (multiplication effects) generally depends on a multitude of sample and counter-specific properties. In real samples there exist normally two main sources of neutrons responsible for induced fission:

- The fission neutrons released from the spontaneous fission of the Pu isotopes to be measured. The probability that these fission neutrons, either as fast neutrons before leaving the sample or as re-entering moderated neutrons (albedo neutrons), cause induced fission increases with increasing sample mass. Furthermore it depends also on a number of additional parameters like geometrical form of the sample, its actinide element and isotopic composition as well as properties of the neutron counter surrounding the sample. The net leakage multiplication is denoted by M_L.
- (α,n) neutrons released from interactions of α particles with light elements in the sample (typically oxygen, boron, fluorine, aluminium or any element with a low atomic number). Again a number of sample properties like density, size, element and isotopic composition, ^{241}Am content etc. are factors determining the strength of this type of neutron source, and hence also the magni-

tude of neutron multiplication due to induced fission. Experience has shown that the presence of fluorine, for example, can lead to gross overestimates (sometimes by a factor of 2 or 3) of the actual amount of plutonium due to neutron multiplication. The ratio of (α,n) to spontaneous fission neutron emission, is denoted by the term α.

The plutonium assay from a neutron coincidence measurement will be also overestimated if other spontaneously fissioning isotopes such as ^{244}Cm are present in the sample. Traces of a few ppm of this isotope in plutonium would lead to (positive) measurement biases in the (few) percent range. Current neutron coincidence counters will not be able to discriminate against this interference at this level.

2.3.2 Instrumentation

The spontaneous fission neutrons emitted by a Pu-bearing sample have an average energy of about 2 MeV. They must be slowed down to thermal energies in order to allow them to be detected with ^{3}He tubes, which are the standard neutron detectors. In practice all passive neutron measurement systems are equipped with neutron moderating assemblies built from moderating materials such as polyethylene, in which the ^{3}He tubes are embedded. Important parameters characterising a neutron counter assembly are the average neutron life-time (die-away time τ) and the neutron detection efficiency (ε) in terms of their magnitude and uniformity. Generally, a larger number of ^{3}He tubes (sometimes more than 100) are incorporated in a neutron coincidence counter to achieve a high and uniform neutron detection efficiency (in best cases up to 70-80%). A high detection efficiency is important for coincidence counting because the probabilities for detecting dual and triple coincidences are proportional to ε^2 and ε^3 respectively. A short die-away time is important as it minimises the accidental coincidence count rate due to pile-up of random neutrons.

Most of the existing neutron coincidence counters in use for Safeguards applications are thermal neutron counters designed for detecting thermalised neutrons, with a typical die-away time $\tau \cong 50$ μs. Recently, less moderated counter assemblies detecting neutrons before thermalisation (epithermal counters) with a significantly reduced die-away time ($\tau = 22$ μs) have been designed and demonstrated.

Besides their categorisation into thermal and epithermal counters, the passive neutron counters are further differentiated according to the manner in which the neutron detector signals are processed and analysed. The two basic variants used in practice are:

(i) *Passive Neutron Coincidence Counters (PNCC)* providing the total neutron counting rate (denoted as 'Totals' or 'Singles'), and the rate of dual coincidences (denoted as 'Doubles' or 'Reals'), and

(ii) *Passive Neutron Multiplicity Counters (PNMC)* extracting in addition to the 'Singles' and 'Doubles' rates also higher-order coincidence events ('Triples',..).

The most common hardware used so far in the PNCC systems for the extraction of the 'Singles' and 'Doubles' rate from the pulse train produced by the ^{3}He detectors is the 'Shift Register'. In the past this has been the most widely used type of coincidence analyser, and it still represents a good choice for the measurement of smaller amounts of well-characterised product materials like Pu metal or Pu-oxide exhibiting small and predictable neutron multiplication effects as well as low and predictable (α,n) production rates. For impure or inhomogeneous materials, such as scraps or waste, however, where corrections for multiplication, matrix and other effects become significant, the two experimental rates (Totals and Reals) are not sufficient for a reliable and accurate Pu assay.

To overcome these difficulties, the passive neutron multiplicity counting technique (PNMC) is being increasingly applied in recent years [50]. The enlarged experimental information provided by PNMC systems (3 measured quantities: Totals, Doubles and Triples) allows extracting quantitative information on existing neutron multiplication effects from the measurement data. In this way systematic measurement uncertainties can be greatly reduced, thus making the plutonium assay by PNMC much more reliable and accurate, particularly for poorly characterised and impure materials with a large induced fission component. For the assay of large containers, for example waste containers, the use of PNMC, already well advanced [51], still can provide improvements. Interpretation of PNMC results for waste assay applications therefore requires considerable expertise and caution, although PNMC has shown to improve the overall accuracy under certain conditions.

2.3.3. Recent Improvements

The Research and development work for improved PNCC and PNMC techniques are still continuing. Some recent advances and improvements, which have resulted in notable improvements in measurement performance for certain applications, have been achieved in the areas of:

- interpretation and assessment of measurement results,
- detector technology,
- undamental nuclear data.

Interpretation:

The fact that a number of specific sample properties is affecting the measured coincidence rate of a neutron coincidence counter poses problems not only for the calibration of neutron coincidence counters, but also for the correct interpretation of the coincidence response in terms of a true value for m_{240eff}. In order to alleviate this situation, predictive modelling methods based on Monte Carlo calculations (mainly with the code MCNP at the present time) are being increasingly employed for the prediction of the neutron response for given samples under well-defined conditions. These calculations are now possible, at least on a relative scale, with very high precision and accuracy. In this manner the number of (expensive) physical standards for calibration can be greatly reduced, and systematic uncertainties could be lowered in favourable cases to ≤ 0.2%.

Closely linked to the issue of measurement interpretation is the increased use of multiplicity counting, which can offer significantly improved reliability for the correct interpretation of the measured neutron coincidence responses in terms of m_{240eff}.

Detector technology:

A new generation of neutron assay counters for a wide range of plutonium items has been

Table IX: Experimental coefficients for the calculation of m_{240eff} from Doubles (D) and Triples (T) coincidences. Values in brackets are 1-sigma uncertainties in % [48, 49].

Counter [a]	γ_{238}(D)	γ_{242}(D)	γ_{238}(T)	γ_{242}(T)
		Counter-specific effective γ-coefficients		
OSL-INVS (ε = 40.3%)	2.707	1.658	-	-
GW = 64 μs	(0.41%)	(0.30%)	-	-
GW = 128 μs	2.714	1.667	-	-
	(0.41%)	(0.30%)	-	-
HENCC95 (ε = 44.3%)	2.7002	1.6586	2.6773	1.6348
GW = 88 μs	(0.88%)	(0.28%)	(1.88%)	(0.95%)
Previously used effective values	2.52	1.68	-	-
	Nuclear data-related component of γ-coefficient			
New experimental values	2.784	1.633	2.803	1.597
	(1.51%)	(1.24%)	(2.63%)	(2.05%)
Values derived from published nuclear data /18/	2.573	1.708	2.645	1.744
	(4.37%)	(2.21%)	(13.8%)	(3.47%)

a) ε = detection efficiency for ^{240}Pu spontaneous fission neutrons
GW = width of coincidence inspection interval

recently introduced with the Epithermal Neutron Multiplicity Counter (ENMC) [47]. The high efficiency of this counter, combined with the significantly reduced neutron die-away time (τ = 22 μs), offers substantial improvement in precision and accuracy especially for problem cases presented, for example, by impure materials with a large (α,n) contribution to the neutron emission.

Fundamental data:

The quantity m_{240eff} as primarily measured by PNCC and PNMC counting is linked to the amount of the neutron emitting Pu-isotopes through the coefficients $_{\gamma 238}$ and $_{\gamma 242}$. For high-accuracy analytical PNCC measurements, and for calibration procedures based, for example, on mono-isotopic ^{240}Pu reference samples, the uncertainty associated with these coefficients should be limited in order not to contribute a significant additional source of systematic uncertainty. The γ-coefficients are determined both by nuclear data-related components (specific neutron emission rates, second moment of the neutron multiplicity distribution for double coincidences and third moment for triple coincidences [47b]), and by a detector-specific component described by the relative difference in detector efficiency for the different average neutron energies of the spontaneous fission neutrons from ^{238}Pu, ^{240}Pu and ^{242}Pu. In a strict sense effective γ-coefficients are therefore counter-specific, but are expected in practice not to change significantly from counter to counter as long as the energy dependence of the counter efficiencies are similar. Effective γ-coefficients obtained for two different counters from recent experimental investigations [48,49], together with the general nuclear data related component obtained after factoring out the effect of detector efficiency, are summarised in Table IX. The new values, with significantly reduced uncertainty, have notably changed compared to previously adopted values.

2.3.4 Performance Values for Passive Neutron Measurements

PNCC is applicable to practically all kinds of Pu-bearing materials. The majority of the PNCC measurements for Safeguards are carried out on relatively pure and well-descript feed and product materials such as Pu-metal, Pu-oxides and MOX materials. The amount of plutonium contained in this type of measurement sample can typically range from the gram level up to several kilograms. A second type of items falling into the category of product materials includes finished physical products like individual MOX fuel pins up to complete MOX fuel assemblies. Accordingly, a large variety of different neutron coincidence counter assemblies have been designed and optimised for the respective applications, including, for example, counters known as Inventory Sample Counter (INVS) for relatively small samples, various types of High Level Neutron Coincidence Counters (HLNCC) with sample cavities accommodating items up to several kilograms of Pu, and counter assemblies for fuel assemblies like the Universal Fast Breeder Counter (UFBC) or the Neutron Coincidence Collar (NCC).

Another important category of materials covered by PNCC includes all kinds of poorly characterised Pu-bearing residues, scrap and waste materials to be assayed both for nuclear management and control, for accountability and for Safeguards verification. Neutron multiplicity counting is becoming the preferred neutron technique for this kind of materials.

During the past years PNCC has been also established as an analytical technique for Pu-element assay on small samples under well-controlled laboratory conditions. In those applications INVS-type counters incorporating also a HpGe detector for simultaneous Pu-isotopic measurements are operated as fixed installations at glove-boxes.

Updated performance values available today for the determination of m_{240eff} from passive measurements based on the shift register technique, thermal neutron multiplicity counting and epithermal neutron multiplicity counting are presented in Tables X, XI and XII. The data given for epithermal multiplicity counting should be considered as purely indicative, since they are based on measurements with a single prototype and therefore ENMC cannot be considered as a consolidated routine technique.

The performance data given in the tables are grouped for the various types of instruments and/or measurement items. The materials are further categorised according to their α-ratio ((α,n)/SF ratio), which represents an important parameter for PNCC measurements. Since counting stati-

stics, and hence the counting time, in many cases is a leading factor determining the magnitude of the random uncertainty component, typical counting times are also given in the tables. This allows scaling the random uncertainty accordingly for different choices of counting times.

The error sources contributing to the given random and systematic uncertainties are manifold. For the PNCC technique the major uncertainty components are due to:

- Counting statistics (random)
- Calibration parameters and uncertainties in reference materials (systematic)
- Correction for multiplication effects, dead time, etc… (systematic)
- Nuclear data.

In PNMC we find the same components of uncertainty as in PNCC, but with certain additional effects and parameters becoming more important:

- Longer counting times are critical to obtain good counting statistics especially for the determination of the 'Triples' (random).
- The correction of dead time effects on the different counting rates (singles, doubles and triples) is difficult to quantify accurately (systematic) [43,44].
- The details of the interpretation model (superfission concept, point model, single exponential decay for neutron live time, etc.) used to solve the problem needs careful assessment, since the assumptions and approximations inherent in the model are critical to the accuracy of the result (systematic).
- Nuclear data have a high importance because of the involvement of the 3rd moment of the neutron emission distribution.

The total random and systematic uncertainties for the determination of the amount of plutonium from a PNCC or PNMC measurement is obtained from a combination of the respective uncertainty components for m240eff as listed in Tables X, XI and XII, and for the quantity $^{240}Pu_{eff}$ as given in Table III in Section 2.1. This yields the relative combined standard uncertainty for the Pu element assay from a combined PNCC and isotopic measurement in accordance with the ISO definition for total uncertainty.

Table X: Performance values for m_{240eff} measured in thermal passive neutron coincidence counters with shift registers.

Type of Counter	Nuclear Material Category	Pu Mass (g)	Counting Time (s)	r (%)	s (%)	Notes
HLNC	Pu Metal	10^2 ~ 10^3	1000	0.5	1 – 2	
(ε = 17%)	PuO_2	10^2 ~ 10^3	1000	0.3	1 – 3	
	PuO_2	1	10 000	0.5	0.5	
	MOX Powders	10^2 ~ 10^3	1000	0.3	3 – 5	
UFBC	PuO_2 Powders	10^2 ~ 10^3	1000	0.5	1 – 2	
(ε = 7%)	MOX Fuel Pin Trays			< 1	2 – 3	
	FBR Fuel Elements			< 1	1 – 2	
INVS	PuO_2	1	10 000	0.2	0.2	a/b/
(ε = 40%)	LWR-MOX	1	10 000	0.2	0.2	a/b/
	FBR-MOX	1	10 000	0.2	0.2	a/b/
NCC-passive	LWR-MOX & FBR					
(ε = 13%)	Fuel Elements		1000	1	1 – 3	

Table XI: Performance values for m_{240eff} measured in thermal neutron multiplicity counting mode [50]

Material Category	SNM Mass (g)	(α,n) /SF rate	Counting Time (s)	r (%)	s (%)
Pu-Metal	2000 2000 4000 200-4000	0 to 0.2 0 to 0.2 0 to 0.2 0 to 1.3	1000 3000 1800 1800	7.1 5.1 2.0 3.3	up to 10
Pu-Oxide	2000 1000 1000 4000 1000	1 1 1 1-4 1-4	5000 3000 1800 1800 600	0.7 0.8 2.2 3.0 1 – 3	
Pu-Scrap	100 100-1200	5 1-6	1000 3600	12 4.5	1-5
Plutonium Residues	120 300 20-100 100	13-29 7-34 8-30 5-9	3000 3600 3600 3600	20 18.9 7 8.7	2-10
Plutonium Waste (estimated)	1 1 1	1 5 20	1000 1000 1000	2 10 50	1-2 2-5 5-10
Pu-Oxide in Excess Weapons Materials	1000 1000 4000 4000	1-10 1-8 1-6 1-6	1500 1000 1800 1800	6.0 5.0 4.2 5.8	1
MOX	300	1-2	1000	1 – 2	1 – 3

Table XII: Performance values for m_{240eff} measured with high-efficiency epithermal neutron multiplicity counters (ENMC) [47].

Configuration	Pu Mass (g)	Counting Time (s)	r (%)	s (%)
ENMC (ε = 65%)	< 10 g 10 – 1000 g	1800 1800	0.5 – 3 0.2 – 1	2 – 3 0.5
ENMC/INVS (ε = 80%)	1 g	3600	0.15 – 0.3	1

a/ Calibrated with certified monoisotopic ^{240}Pu reference sample
b/ Neutronic response of reference sample and measurement samples modelled by MCNP calculations

2.4 Gamma/X-Ray/Weighing (GXW) Method for Pu Isotopic Composition and Element Content

A new NDA technique known as Gamma/X-Ray/Weighing (GXW) method for the simultaneous determination of the plutonium element concentration and isotopic composition both in solid and liquid samples from a single HRGS measurement has been developed and tested during the last decade [52,53,54,55]. If combined with a sample weighing, the method determines the total Pu content in a sample.

2.4.1 Principle

The GXW method represents a generalisation of the standard plutonium isotopic measurements by HRGS in that it exploits the full spectroscopic information contained in a gamma spectrum from a plutonium sample to achieve also a determination of the plutonium mass fraction in the assay material. The method makes use of several gamma-spectrometric analysis techniques such as enrichment-meter-type measurements, passive differential gamma absorptiometry (PDGA) and passive X-ray fluorescence analysis (PXRF), either individually or in combination, for the determination of the plutonium element mass fraction from a single gamma spectrum.

The approach works in principle for any kind of plutonium-bearing materials, liquids and solids, provided the recommended measurement geometry is set up. A calibration or normalisation with at least one reference sample is normally required for the mass fraction measurement. The measured Pu mass fraction together with the knowledge of the net sample weight obtained from a weighing yields the total amount of plutonium.

2.4.2 Performance

A number of exercises involving measurements on liquid and solid plutonium materials have been carried out for demonstrating the capabilities and the performance of the method for Pu mass fraction measurements. Accurate measurements require a well-defined and controlled counting geometry, particularly for samples that are not “infinitely” thick.

Table XIII gives some indicative performance data for mass fraction measurements [55,56]. The performance values are given in terms of relative standard deviations for the differences between measured and declared mass fractions observed in demonstration exercises. Most of the measurements on Pu-bearing solid samples have been performed on MOX materials (powders, pellets). Results for those samples showed an agreement with reference values of the order of 1% or better for defined and uniform counting conditions. If the measurements were subject to some variability in the counting geometry, for example due to varying physical or geometrical properties of the measured samples, the accuracy degraded to about 2.5%, pointing to some degree of sensitivity of the method to experimental conditions. The method is still undergoing further optimisation for routine application.

For the isotopic composition measurement the GXW method makes use of the intrinsically calibrated analysis of the XKα region. Pertinent performance data are quoted in Section 2.1.

Table XIII: Performance of the GXW method for Pu mass fraction measurements.

Material type	Range of mass fraction (wt. %)		Typical uncertainty rsd (%)	
	W_U	W_{Pu}	W_U	W_{Pu}
PuO_2		85 – 88		1.5
MOX	20 – 80	2 – 65	2.5	1 - 2

3. TECHNIQUES FOR URANIUM-BEARING SOLID MATERIALS

This section discusses features and performances of the following NDA techniques for measurements on uranium-bearing solid materials:

- Gamma spectrometry for the determination of the ^{235}U enrichment in low-enriched uranium materials such as uranium oxide powders and pellets, fuel pins and assemblies, and UF6 in storage containers;
- Active neutron interrogation for the determination of the ^{235}U content in feed and product uranium materials;
- Gamma/X-ray/weighing method for the simultaneous determination of the ^{235}U enrichment and uranium element content.

3.1 Gamma Spectrometry for the Determination of the ^{235}U Enrichment in Low-Enriched Uranium Materials

The determination of the ^{235}U enrichment in uranium materials by means of gamma spectrometry represents one of the most widely used NDA applications since the beginning of Safeguards verification measurements. The method, which has early matured because of its simplicity, uses two different approaches for the measurement of the ^{235}U enrichment: (i) the “infinite thickness” approach, and (ii) the intrinsic calibration approach.

3.1.1 Infinite Thickness Approach

This approach is based on the so-called enrichment meter principle [57,58]. Here the most prominent gamma ray of 185.7 keV from the decay of ^{235}U is counted in a defined counting geometry from uranium samples satisfying the “infinite thickness” condition, i.e., which must be opaque for 186 keV photons. Under this condition the measured counting rate of 186 keV photons is proportional to the ^{235}U enrichment. The required infinite sample thickness ranges from about 0.25 cm for metal samples to about 7 cm for UF_6 with a density of 1 g cm^{-3}. Small corrections to the measured counting rates are needed to account for different chemical compositions of the uranium samples [58,59]. The method is best suited for bulk samples (e.g., uranium oxides and UF_6 in storage containers), which easily meet the infinite thickness requirement.

The enrichment measurement based on the enrichment meter principle requires physical standards containing a sufficiently large amount of uranium reference materials for calibration. The ESARDA NDAWG has early launched the production of a set of internationally certified calibration standards for this purpose [60,61], and a detailed user’s manual for accurate ^{235}U enrichment measurements using these calibration standards has been issued [59].

3.1.2 Intrinsic Calibration Approach

Other methods based on the so-called intrinsic calibration approach have been developed to circumvent the need for calibration with physical standards. In these methods the $^{235}U/^{238}U$ isotope ratio is determined from the measured gamma spectrum using corresponding gamma and X rays from the decay of both isotopes. Since ^{238}U does not directly emit a useful gamma ray from its decay to ^{234}Th, gamma rays from the decay of its daughter nuclides ^{234}Th and ^{234}Pa must be used instead. This approach requires secular equilibrium between ^{238}U and its daughter nuclides, which is reached about 80 days after chemical separation. The method is therefore not suited for freshly separated uranium materials.

Since a few years the preferred method for intrinsically calibrated enrichment measurements bases on the analysis of the $XK\alpha$ region (89-99 keV), where fairly abundant but strongly overlapping gamma and X-ray signatures from the ^{235}U and ^{238}U daughter nuclides ^{231}Th and ^{234}Th occur [62,63]. This method allows rapid enrichment measurements on arbitrary uranium samples. The principal limitation of the method, besides requiring secular equilibrium for the $^{235,238}U$ daughter

nuclides, arises from the fact that the gamma and X-ray signatures from ^{231}Th and ^{234}Th become strongly unbalanced in intensity at very low (< 1%) and very high (> 90 %) enrichments. This makes the measurements for those enrichment grades less precise and accurate [64].

An alternative but nowadays less frequently used intrinsic calibration approach makes use of the ^{235}U gamma rays ranging from 143 to 205 keV, and of the ^{234}Pa gamma rays occurring between 258 and 1001 keV [65,66,67]. The principal problem associated with this type of analysis is the difficulty to establish a reliable link between the relative detection efficiency for the radiation from both isotopes.

3.1.3 Detectors

The choice of the detector is an important criterion for the performance of a gamma-spectrometric enrichment measurement. For many years the enrichment measurements have been mainly made using either high-resolution HPGe detectors or low-resolution NaI(Tl) scintillation detectors. Both types of detectors can be principally applied for enrichment measurements based on the infinite thickness approach. As a matter of fact, the majority of the in-field measurements in the past used low-resolution scintillation detectors, with some degradation in performance (typically a factor of 2) compared to the results obtained with high-resolution HPGe detectors. For field measurements NaI(Tl) detectors remain a useful option, particularly when used in conjunction with improved software for spectrum analysis [68]. However, NaI(Tl) scintillation detectors cannot be used for enrichment measurements based on intrinsic calibration because of their inherently limited energy resolution.

In recent years semiconductor detectors such as CdZnTe (CZT) or CdTe operating either at room temperature or at slightly reduced temperature levels simply reached by Peltier cooling have become another viable detector alternative for enrichment measurements. Advanced technologies have permitted the fabrication of small volume detectors with good gain and efficiency stability when submitted to temperature variation, high intrinsic efficiency, and good energy resolution (FWHM at 186 keV up to 2.5 keV for room temperature CZT, and up to 1.7 keV for Peltier-cooled CdTe [69]). These detectors are in principle applicable to both the infinite thickness [70] and intrinsic calibration type of enrichment measurement. However, enrichment measurements using CdTe or CZT detectors are still suffering from measurement precision due to the relatively small detector sizes available so far.

For the time being the CZT and CdTe detectors are mainly used for more qualitative attribute measurements. Typical Safeguards applications are commonly related to locating, monitoring or identifying nuclear materials. Recent improvements in the construction technology of larger volume CZT detectors and the use of new methods and algorithms for spectrum analysis have proven that those detectors, as an alternative to the systems based on HPGe and NaI(Tl) detectors, are also suited for routine ^{235}U enrichment measurements in drums, cylinders and other types of containers containing nuclear material. They have also become a useful tool for gamma-ray signature measurements to verify the ^{235}U enrichment in fuel pins [71,72], or even of inner rods within a fuel assembly, where the small detector size makes it possible to insert the detector between rows of rods.

3.1.4 Performance values

The performance values established for the above mentioned methods for gamma-spectrometric enrichment measurements, when applying different detectors and counting times, were mainly derived from an international exercise managed by the ESARDA NDA Working Group [5], and are further based on practical experiences of Working Group members as well as on literature data [64,66,73].

Table XIV lists the performance values for measurements on uranium oxide materials. The achievable measurement performance depends on the type of analysis (infinite thickness or intrinsic calibration), on the type of the detector and on the counting time. The random uncertainties are mostly due to counting statistics and apply for the given counting times. The systematic uncertainties are generally lower for well-calibrated infinite thickness measurements than for intrinsically calibrated measurements. Systematic uncertainties introduced by the unfolding of the complex XKα-region and also the poor knowledge of

some of the nuclear data (gamma emission probabilities) involved in the analysis are the main factors responsible for the poorer performance of the intrinsic calibration approach. The data in the Table also underline the fact that measurements made with CZT detectors cannot yet compete in performance with measurements made with HPGe detectors offering higher efficiency and energy resolution.

Enrichment measurements made on UF_6 in standard storage containers represent a special case. The relatively thick container walls strongly attenuate the radiation in the XKα-region, which practically only leaves the counting of the 186 keV gammas in the infinite geometry for the enrichment measurements. Further, because of the relatively poor signal to background ratios obtained from the very large and well-shielded measurement items, reasonable measurements are only possible with high-resolution HPGe detectors.

Table XV lists performance values that can be obtained under carefully fixed experimental conditions [73]. The random uncertainty components mainly depend on the counting time and the geometrical set up, including a possible collimator. For LEU in 30″ containers, the random uncertainty typically ranges between 1.5% (counting time of 3000 s) and 3% (1500 s). For natural or depleted UF_6 in 48" containers, the random uncertainty is about 7%.

Systematic uncertainty components for this type of enrichment measurements are determined by a number of factors like the poorly defined physical properties of the UF_6 (liquid, solid, spatial inhomogeneity), attenuation corrections to be made for the container walls, interfering radiation from deposits in the inner wall of the containers and from neighbouring storage containers, and the calibration procedure. They can vary from 2 to 10 % depending on the actual measurement conditions, on the evaluation and measurement procedures used, and on the calibration standards available for calibration.

One of the major problems for this kind of samples is due to the wide variability of the physical properties of UF_6 in storage conditions. Efforts to apply passive or active neutron measurement for uranium enrichment determination in UF_6 are presently ongoing.

Table XIV: Performance values for gamma-spectrometric enrichment measurements on low-enriched uranium oxide materials.

^{235}U Enr.	Infinite thickness method									Intrinsic calibration method					
	HRGS (Ge detectors)			LRGS (NaI detectors)			CZT			HRGS (Ge detectors)			CZT		
	CT (s)	r (%)	s (%)	CT s	r (%)	s (%)	CT s	r (%)	s (%)	CT s	r (%)	s (%)	CT s	r (%)	s (%)
0.3 to 0.7%	360	2	1	360	3	1	1200	10	1	360 3600	8 3	5 5	ns	ns	ns
2 to 4 %	360	0.7	0.5	360	1	0.5	1200	3	1	360 3600	2 1	1 1	104	10	5
5 to 10 %	360	0.5	0.5	360	0.5	0.5	1200	3	1	360 3600	2 1	1 1	104	10	5

ns = not suitable

Table XV: Performance values for enrichment measurements on UF_6 in storage containers.

Sample type	r (%)	s (%)
LEU	1.5 to 3	2 to 10
Natural or depleted U	7	2 to 10

3.2 Active Neutron Techniques for the Determination of the ^{235}U Content in Uranium Feed and Product Materials

3.2.1 Principles of Active Neutron Coincidence Counting

Due to the very low spontaneous fission yields of all the uranium isotopes, passive neutron coincidence techniques are generally not suitable for the assay of uranium bearing samples (an exception is the use of (alpha,n) reactions from ^{234}U in uranium fluoride). However the fissile content in a sample can be readily measured by adding an external interrogation neutron source. The neutrons from the interrogation source will induce fission in the fissile nuclei of the sample. Neutron induced fission (like spontaneous fission) results in the simultaneous emission of several prompt neutrons ($<\nu>=2.41$ for fission induced by thermal neutrons in ^{235}U). The coincidence counting technique allows the distinction between events with the emission of single or multiple prompt fission neutrons. This makes it possible to discriminate between neutrons from the primary interrogating source and those from fission induced in the sample, provided that the primary source generates randomly non-correlated single neutrons. Coincidence counters with a random interrogation source are known as Active Neutron Coincidence Counters (ANCC).

Among the radioactive sources those based on (α,n) reactions are t he best candidate for active neutron interrogation. A frequently used source is AmLi. The main advantage of the AmLi source with respect to other (α,n) reactions is the low energy of the emitted neutrons: the mean energy is 0.54 MeV, which minimises the probability of fast fission in ^{238}U.

For small samples the “Reals” coincidence rate is proportional to the quantity of fissile material in the sample. For large samples the self-shielding phenomena limit the “visibility” of fissile material to the interrogating neutrons, causing saturation effects in the response function and underestimation in the quantity of the fissile material (unless the calibration is designed to take the effect into account). This self-shielding effect is one of the major contributors to the systematic assay error of active neutron techniques.

3.2.2 Active Neutron Coincidence Counting Instruments

Apart from the presence of the interrogating source, the methods and procedures of shift-register based instruments for active neutron coincidence counting are very similar to those used in PNCC counting.

There are basically two major families of instruments in this category:

- the Neutron Coincidence Collar (NCC) in active mode;
- the Active Well Coincidence Counter (AWCC).

Neutron collars are typically composed of four slab detectors in a square arrangement, and are used for the assay of fresh fuel assemblies. Some models have a modular layout allowing the adjustment of collar dimensions to the fuel element size, others have fixed configurations for specific fuel type (PWR and BWR). Collars can be used both in passive and active mode. For passive only applications (MOX fuels) normally all the four sides are equipped with detectors, for active/passive applications (LEU fuels) only three detection slabs are used and the fourth wall hosts the source.

Active well coincidence counters are general-purpose devices for uranium bearing samples at practically any enrichment (HEU and LEU), chemical form (metal, oxide) and physical form (powders, pellets, plates, MTR elements). An AWCC is conceptually similar to a passive HLNCC except for the presence of two AmLi sources in the top and bottom polyethylene plugs. It can be operated either with or without a cadmium liner (fast or thermal mode).

3.2.3 Active Neutron Coincidence Counting in Multiplicity Mode

By extending the shift register electronics it is possible to operate ANCC systems in multiplicity mode. This is exactly analogous to the extension from PNCC to PNMC. Under certain conditions three unknown quantities can then be determined instead of just two. This allows, for example, a

variable detection efficiency (perhaps due to variable moisture content) to be taken into account in the interpretation model. The use of multiplicity counting in ANCC systems is still undergoing development, and therefore no performance values are available to be quoted in this document.

3.2.4 Performance Values for Active Neutron Coincidence Counting

Performance values for the assay of the fissile uranium content obtained with two common instruments (NCC and AWCC) from different materials are given in Tables XVI and XVII, essentially based on field experiences [74,75]. Note that these values assume that a representative calibration exists, for each material type quoted. The systematic uncertainty for the fast mode assay is generally higher than for the thermal mode, due to the range of matrix effects, although the potential for gross assay underestimation is greatly reduced in fast mode.

It is worth noting that the AWCC technique is also used for other purposes, including waste characterisation. For these applications, where the physical composition of the waste package and fissile material can vary considerably, the assay uncertainties can be much greater than the figures quoted below.

Table XVI: Performance values for the determination of the ^{235}U mass loading in fresh LEU fuel elements (1000 s counting time).

Technique	Objects	Enrichm.	r(%)	s(%)
NCC (active mode)	UO_2 Fuel Elements for LWR	Up to 3%	1	1 – 2
	UO_2 Fuel Elements for LWR	3 – 5 %	1	2 – 4
	LWR fuels with burnable poisons	any	1	3 – 5

Table XVII: Performance values for the determination of the fissile content in U samples.

Technique	Objects	r(%)	s(%)
AWCC	HEU Metal	2	3
	HEU Powder (fast mode)	2	10
	HEU Powder (thermal mode)	2	5
	UF_4 Salt	5	2
	HEU/Th/C Pebbles	2	4
	HEU/Al MTR	1	3
	LEU Powder (fast mode)	2	5

3.2.5 Active Total Neutron Counting

Uranium bearing samples can be assayed through active neutron interrogation e.g. by PHONID (PHOto-Neutron Interrogation Device) [76] followed by total fission neutron counting.

The PHONID device contains two photo-neutron (^{124}Sb,Be) sources. The energy of these neutrons is relatively low (up to 400 keV), below the threshold of fast fission in ^{238}U, so thermal fission will be induced in ^{235}U only. Fission neutrons are then detected by an array of ^{4}He detectors for fast neutrons. Neutron counting is made in “Totals” mode.

In the PHONID system the discrimination between source and induced fission is not performed

Table XVIII: Performance values for total neutron counting active technique on U-bearing materials

Technique	Objects	r (%)	s (%)
PHONID	HEU-MTR Platelets	1	2
	HEU-Metal	1	1
	LEU-UO_2 Powder	1	1.5
	LEU-U_3O_8 Powders	1	2.5

by the electronics like in the coincidence counters, but it relies on the sensitivity of ^{4}He detectors to the neutron energy which is much higher for fast neutrons than for thermal ones.

Performance values for the PHONID technique are shown in Table XVIII. The random components of the uncertainty are mostly determined by counting rates and counting times, background subtraction and normalisation. Systematic components are linked with neutron multiplication, calibration curve set up, accuracy of calibration standards, moisture and impurities. These values assume that a truely representative calibration exists, for each material type quoted.

Although PHONID proved to be a reliable and effective technique in measuring bulk samples of uranium bearing material, it did not encounter the favour of analysts since the procedure for its application is quite cumbersome. One of the main drawbacks of the technique is its high sensitivity to a lot of parameters affecting the measurement. For example, changes in the geometry of the sample, container and packaging material, matrix, and enrichment, all cause an alteration of the response function, requiring an extensive calibration effort. Practically any family of similar samples requires a separate calibration. Moreover the short half-life of the ^{124}Sb (60 days) requires frequent, and costly, replacements of the source.

For all these reasons, recently feasibility studies have been started to replace PHONID with passive techniques. In principle the amount of uranium can be measured by passive neutron coincidence counting in the same way as for plutonium, since its isotopes (principally ^{238}U) also decay by spontaneous fission. Therefore, through a combination of a neutron measurement with gamma spectrometry for enrichment, it becomes possible to verify uranium samples. A substantial problem comes from the low specific neutron yield of uranium due to the very long half-life. Counters with high efficiency are therefore required. Studies have shown that a statistical measurement uncertainty (below 0.5%) can be reached in a reasonably short counting time (1000 s) for large samples (uranium mass from 100 g to several kg) provided the counter has an efficiency of the order of 50% [77]. Design, construction and testing of a passive neutron device for large LEU samples are currently ongoing. Use of PNCC counting, in this way, is an established technique in NCC collars for ^{238}U assay of fresh fuel assemblies.

3.3 Gamma/X-Ray/Weighing (GXW) Method for the Determination of Enrichment and Uranium Mass Fraction

The GXW method described in Section 2.4 also provides measurement capabilities for the determination of the ^{235}U enrichment and the uranium mass fraction [55]. The underlying enrichment measurements are either based on the infinite thickness approach or on the intrinsic calibration approach as described in Section 3.1, and the performance values reported in Table XIV also apply here.

3.3.1 Principle of the Mass Fraction Determination

The method applied for the uranium mass fraction measurement depends on the type of detector used. For measurements made with HPGe detectors the uranium mass fraction is determined either by means of the Passive X-Ray Fluorescence (PXRF) technique, or by means of the Passi-

ve Differential Gamma Absorptiometry (PDGA) technique, or from a combination of both [55]. The PXRF method relates the measured intensity ratio of passively excited UKα X rays and 186 kev gamma rays from ^{235}U through a polynomial function to the uranium mass fraction. In the PDGA method the measured intensity ratio of isotopic gamma rays from ^{235}U (144/186 kev ratio) is related to the uranium mass fraction. Both methods require a calibration to establish the respective functional relationship between measured intensity ratio and uranium mass fraction.

In measurements made with NaI detectors the ^{235}U enrichment and the uranium mass fraction are determined in an iterative process from the measured intensity ratio of 186 keV (^{235}U) and 1001 keV (^{234}Pa) gamma rays.

3.3.2 Performance Values

Typical performance values for the uranium mass fraction determination in solid uranium samples by the GXW methods, expressed in terms of relative standard deviations for the differences between declared and measured values, are given in Table XIX. The data are based on recent performance assessments [55,56]. For measurements made with a HPGe detector an accuracy level of about 1% relative has been achieved. For measurements made with a NaI detector only few data are available, but an accuracy level of 1.5 % seems to be reachable.

Table XIX: Performance values for uranium mass fraction measurements in solid U-bearing samples by the GXW method.

Detector	Range of U mass fraction (wt. %)	Typical uncertainty rsd (%)
Ge	5 - 85	1.0
NaI	5 - 85	1.5

4. TECHNIQUES FOR U AND PU IN LIQUID FORM

The prime quantity usually determined by NDA techniques from nuclear samples in liquid form is the uranium and/or plutonium element concentration. The major technique currently applied for this purpose is K-Edge Densitometry (KED), which is ideally suited for concentration measurements at elevated concentration levels (≥ 50 g/l) in any type of solutions, containing both unirradiated and irradiated nuclear materials. For lower concentrations the technique of KED can be complemented by the technique of X-Ray-Fluorescence (XRF) analysis, which offers a larger dynamic range at the expense of slightly inferior overall measurement accuracy compared to KED.

Gamma spectrometry is also frequently used for uranium and plutonium concentration measurements. This technique also allows the simultaneous measurement the isotopic composition of plutonium and/or the ^{235}U enrichment of uranium. A direct gamma measurement, however, is only possible for unirradiated materials, while the analysis of irradiated materials requires a prior separation of fission products.

Radiometric techniques such as KED, XRF and HRGS are nowadays also frequently utilised in analytical measurements for the determination of element concentrations in solid samples such as powders and pellets. In those applications the solid samples need to be dissolved prior to the radiometric measurements. This is therefore no longer a purely non-destructive analysis. Nonetheless, we will also quote performance values for those applications, because the radiometric measurements applied to the dissolved samples are the same as those used for original liquid samples.

4.1 K-Edge Densitometry and XRF for Element Concentration

4.1.1 Principles

K-edge densitometry is a special form of photon absorptiometry [78,79]. In KED the photon transmission through the sample under assay is measured at two photon energies which should bracket

as closely as possible the K-absorption edge energy of the element of interest. The logarithmic ratio of the photon transmission measured below and above the absorption edge is directly proportional to the volume concentration of the measured element in the liquid and to the thickness of the investigated solution layer. The latter parameter requires sample containers of well-defined path length holding the sample solution for the transmission measurements. KED provides an element-specific measurement because the K-absorption edge energy represents an element-specific signature.

KED is one of the most accurate NDA techniques because it reduces the determination of the quantity to be measured, i. e. the volume concentration of the element of interest, to a simple ratio measurement. This helps to significantly reduce or even eliminate a number of systematic uncertainty components present in other NDA measurements. The technique is even able to determine the concentration, at a reduced accuracy of about 2%, without any calibration, if the measurement evaluation is based on physical constants alone. For higher accuracy the KED measurements are usually calibrated against certified reference solutions. The lower concentration limit for accurate KED measurements is normally set at about 50 g/l.

The technique of XRF has to be applied if the element concentration to be determined falls below the useful range for KED measurements. The energy-dispersive analysis of fluoresced K-X rays is applicable for quantitative concentration measurements down to concentration levels of about 0.5 g/l, with detection limits settled in the range of 0.02-0.05 g/l. The interpretation of the XRF measurements is not as straightforward as for KED, and it is also more sensitive to matrix effects. However, the XRF technique becomes very accurate for element ratio measurements to determine the concentration of a minor element relative to the concentration of a major element known from a KED measurement. The simultaneous determination of the U and Pu concentration in reprocessing input solutions, with a typical U/Pu-ratio of ≅ 100, represents an important example for such an application [80,81].

4.1.2 Instrumentation

The majority of the K-edge densitometers used for Safeguards are equipped with an X-ray generator as photon source for the transmission measurement [82]. This offers enough flexibility to tune the measurement conditions to the requirements of different applications. The high photon strength provided by an X-ray tube also makes measurements on highly radioactive samples possible. It can be also used as excitation source for simultaneous XRF measurements. The Hybrid K-Edge Densitometer (HKED) combining KED and XRF has become a standard technique and instrument for Safeguards verification measurements in reprocessing plants [83]. All K-edge densitometers equipped with an X-ray generator are stationary instruments attached to shielded or unshielded glove-boxes for sample handling.

Another type of K-edge densitometer applied to the analysis of uranium samples uses radioactive isotopes as photon source. A common densitometer of this type is the COMPUCEA instrument [84,85]. The currently adopted isotopic source for the K-edge measurements on uranium samples is a mixed $^{57}Co/^{153}Gd$ source. The COMPUCEA instrument is mostly used as a mobile instrument during physical inventories in fuel fabrication plants [86].

Table XX: Performance values for volume concentration in liquid samples from direct KED/XRF measurements.

Type of Sample	Technique	Measurand	Counting Time (s)	r (%)	s (%)	Remark
U-nitrate	KED	U-conc.	2000	0.2	0.15	> 100 g/l
U-nitrate	COMPUCEA	U-conc.	2000	0.2	0.15	> 100 g/l
U-nitrate	XRF	U-conc.	2000	0.5-0.2	1	1-50 g/l
Pu-nitrate	KED	Pu-conc	2000	0.2	0.15	> 100 g/l
Pu-nitrate	XRF	Pu-conc	2000	0.5-0.2	1	1-50 g/l
Reproc. input solution	HKED	U-conc.	2000	0.2	0.15	150–250 g/l
		Pu-conc.	3600	0.6	0.3	1-2 g/l

Table XXI: Performance values for element concentration in solid samples (powders, pellets) from KED/XRF measurements after sample dissolution (typical sample size ≅ 2 g).

Type of Sample	Technique	Measurand	Counting Time (s)	r (%)	s (%)	Remark
Pu-oxide	KED	wt. % Pu	2000	0.2	0.2	
Pu-oxide	XRF	wt. % Pu	2000	0.2	0.15	Internal U-spike
MOX	HKED	wt. % U	2000	0.2	0.2	
		wt. % Pu	2000	0.3	0.2	
U-oxide	COMPUCEA	wt. % U	2000	0.2	0.2	
	KED	wt. % U	2000	0.2	0.2	

4.1.3 Performance Values

Performance values for the determination of the uranium and plutonium volume concentration in liquid samples by means of KED and XRF are listed in Table XX (see also [30]). The main component dominating the random uncertainty is counting statistics, whereas short and long-term instrument variability and the uncertainties associated with reference solutions used for calibration represent the major error sources contributing to the systematic uncertainty. The application of strict procedures for measurement control and assurance are vital to keep the systematic uncertainty at the quoted level.

Performance values for the determination of the percentage element content in solid samples, obtained from KED/XRF measurements after sample dissolution, are listed in Table XXI. The listed random and systematic uncertainty components include also error contributions from sample weighing, sample dissolution and from the additional solution density measurement needed for the conversion of the volume concentration measured from the dissolved sample into the wt. % concentration of the respective element in the original solid sample.

4.2 Gamma Spectrometry for Element Concentration and Isotope Abundances

4.2.1 Principles

The uranium and plutonium element concentration in liquids can be determined from a purely passive HRGS measurement. The underlying measurements are based on intensity measurements of X-rays and isotope-specific gamma rays from uranium and/or plutonium isotopes in a well-defined counting geometry. Element mass fractions are determined via the PXRF and PDGA techniques, which are part of the general GXW method mentioned in Sections 2.4 and 3.3. For low concentration levels photon counting of isotope-specific gamma rays in a well-defined and calibrated counting configuration represents the preferred measurement approach.

The interpretation of the passive mass fraction measurements in terms of a total mass fraction of the analyte requires the knowledge of the isotopic composition of the uranium and/or plutonium material under assay, which can be derived from the same HRGS measurement as well. Plutonium isotope abundances are either determined through the intrinsic calibration approach as described for Pu solid material, or through calibrated photon counting of isotope-specific gamma rays. For the ^{235}U enrichment measurement from uranium solutions calibrated 186 keV photon counting is the preferred method. In the COMPUCEA instrument this counting is made in a HPGe well counter on a defined volume of uranium solution with known uranium concentration obtained from a parallel KED measurement [84,85].

Table XXII: Performance values for the determination of element mass fractions and isotope abundances in solutions.

Solution	Technique	Measurand	Range	r (%)	s (%)
Pu	GXW	Pu mass fraction	0.2-13 wt%	1	1
U	GXW	U mass fraction	15-30 wt%	2.5	1
	GXW	^{235}U enrichment	0.7-90%	1.5	1.5
	COMPUCEA	^{235}U enrichment	2-4%	1.5	1.5

4.2.2 Performance Values

Performance values for the mass fraction and isotope abundance measurements in solutions by means of HRGS are listed in Table XXII [30,55,56,86]. Performance values for Pu isotope abundance measurements are not quoted, because the most simple and reliable Pu isotopic measurement is made using the intrinsic calibration approach, yielding for liquid samples comparable performance as for solid Pu samples (Table I).

4.3 Other techniques

The techniques of calorimetry and PNCC, mostly applied for solid Pu-bearing samples, can be also used in principle for the determination of the amount of plutonium contained in a liquid sample, provided the amount of plutonium is large enough (≥ 1 g) to allow a precise assay by either technique. Performance values given in Sections 2.2 and 2.3 for solid samples will then also apply for liquid samples.

5. TECHNIQUES FOR SPENT FUEL ASSAY

5.1 Summary of the NDA techniques for spent fuel assay

NDA techniques on spent fuels are extensively used in safeguards both to verify items with regards to their attributes (level of irradiation, cooling time, fuel nature: MOX, LEU, HEU) and, with higher accuracy levels, to quantitatively determine the amounts of nuclear material contained: this second level of NDA would essentially contribute to the nuclear material accountancy in strategic MBAs. Quantitative assay requires the determination of the amount of nuclear materials (NM) through the evaluation of the spent fuel burnup, or a direct NM assay. The burnup determination is generally obtained with passive NDA techniques, also in case when direct nuclear material assessment in spent fuel requires active NDA techniques. At the moment passive techniques are under routine use. Active methods, that have been extensively studied, are less used due to their cost and complexity.

For the monitoring of irradiated spent fuel assemblies, four non-destructive methods can be applied:

- gamma spectrometry using HRGS or RTGS,
- total gamma counting with ionisation chambers,
- passive neutron counting using fission chambers,
- active neutron counting with isotopic neutron sources or neutron generators.

These methods allow the determination of the main physical parameters of an irradiated fuel assembly such as burn-up, cooling time and effective multiplying factor (k_{eff}). The evaluated performances as far as the above parameters are concerned are shown in Ref. [89].

The passive and active neutron signals from a spent fuel assembly can be used to evaluate the

U-Pu residual mass:

- Passive neutron counting is well correlated with the Plutonium mass via the Curium passive neutron emission, which is actually assayed.
- Active neutron counting gives the fissile mass content ($^{235}U + ^{239}Pu + ^{241}Pu$) and, using passive neutron counting, leads to the ^{235}U mass content of the assembly [90,91].

In practice NDA on spent fuel for safeguards purposes, is often performed in reactor ponds and in wet or dry storage. As a consequence, devices used for spent fuel safeguards have to consider underwater operation in hazardous areas. In addition, instrumentation has to cope with very high radiation dose background, so requiring important shielding.

Performance values for the quantitative assay of Pu and U in spent fuel are mostly based on laboratory experiment (Table XXIII); safeguards verification activities are generally attribute verifications (eg burnup determination: Tables XXIV, XXV): the gap evidenced between the performances of the two areas reflects the difficulties encountered in transporting techniques from R&D to field, for the reasons outlined before: complexity of the field conditions, costs of the equipment, transportability of the equipment etc.

When safeguards measurement on spent fuels are attribute tests or semi-quantitative assay, it is more difficult to associate a performance value. However, information on routine in-field measurements is available for the same techniques applied for instance, in the frame of criticality safety [87]. It appears therefore appropriate to list here devices and related performances that are not actually used yet in the frame of safeguards, but could be applicable in future, together with performances of devices which are extensively used for safeguards and other purposes (the FORK device [93, 94, 95] is an example).

5.2 Performances of Uranium-Plutonium Mass evaluation in spent fuels

The random and systematic uncertainties related to the Plutonium mass determination are presented in Table XXIII. The figures for Pu mass estimation results from the use of an on-line depletion code, which calculates the Pu mass based on the declared irradiation history.

Another method is based on direct correlation between the neutron emission and the Pu mass [88]. This method does not use any on-line code and delivers consistent results on the total Pu balance for batches of several assemblies.

Table XXIII: Performance values related to the uranium-plutonium mass evaluation of a spent fuel assembly

Technique Used	Calibration	Physical Parameter Determined	r(%) (*)	s (%)
Passive neutron counting + on line depletion code	No (only detector yield)	Plutonium mass [87]	0.3 - 1.3	1. - 2.5
Passive neutron counting + correlation law (NE=f(Pu))	No (only detector yield)	Plutonium and Uranium mass [88]	0.3 - 1.3	(**) (PWR) 1. (BWR) 2.
Active neutron interrogation	Yes	Fissile mass $^{235}U+^{239}Pu+ ^{241}Pu$ [90]	0.15 - 2.5	< 4.

(*) These values depend on the kind of assembly, burn-up, initial enrichment and the irradiation history.

(**) These values are related to total balance of Pu for batches of several assemblies.

5.3 Determination of physical parameters of irradiated fuel assemblies

Knowledge of the physical parameters of irradiated nuclear fuel is required both for safeguards and operational safety purposes. Measurements on spent fuel encounter “logistic” difficulties as above mentioned, but also principle difficulties, as for instance the lack of primary physical standards, well calibrated and certified [87]. Instruments may be calibrated for gamma ray efficiency, efficiency to thermal neutrons etc., but these parameters may be used only for the quality assurance of the instrumentation and of the methods and not for calibrating for quantitative assay.

Calibration for NDA measurements on irradiated fuel is performed against some set of characterised assemblies (referred as Measured Calibration Curve mode in the Tables) and then used for later measurements [93]. Other methods use on line depletion codes [92, 96, 97] to establish the correlation link (Calculated Correlation Curve) between emissions and physical parameters: in this particular case the detector yields have to be determined by a reference measurement or calculation.

The importance of such measurements stimulated the development of a large variety of instruments and applications of known instruments to irradiated fuel. As for many other instruments used for safeguards purposes, a distinction should be made between mobile devices and fixed installations (in reprocessing plants, reactors...). Fixed devices are usually better calibrated and maintained, some have better characteristics as shown in **Table XXVI**.

National and international inspectors in field mostly use mobile devices. Fixed devices are usually used at reprocessing plants as a part of technological process or combined with Containment and Surveillance systems.

All instruments determine some irradiation signature for the assessment of unknown parameters by the method explained in Ref. [87].

Tables XXIV and XXV present performance values for mobile instruments based on neutron and combined neutron-gamma methods respectively. Some devices, presented as examples, are not intended for safeguards but used in the frame of criticality control. Nevertheless they are of potential interest for safeguards because they are transportable and could be used for in-field applications.

Systematic uncertainties on burnup, cooling time and Pu mass strongly depend on the number and the quality of input data used for signal interpretation. It is to be noticed that the active neutron

Table XXIV: Performances of neutron-only NDA for Spent Fuel: underwater mobile devices.

NDA Method	Physical parameter assessed	Example of devices (min)	Assay Duration	Calibration interpretation	r(%)	s (%)
Relative Passive Neutron	**LEU** Average BU MOX to LEU distinction	FORK Safeguards Safety - criticality [93]	10	Measured Correlation Curve(1)	1	10
Absolute Passive Neutron	**LEU** Average BU Extremity BU Pu amounts	PYTHON Safety - criticality	10	Calculated Correlation Curve(2)	0.1	2 - 5
Active Neutron	**MOX** Average BU Pu amounts K_{eff}	PYTHON Safety- criticality	10	Calculated Correlation Curve: K_{eff}=f(Active Count Rate)	1	2
Passive neutron	**LEU** Average BU Pu amounts	SMOPY [101] Safeguards	10	Calculated Correlation Curve	1	5

Table XXV: Performances of gamma and (gamma + neutron) NDA for Spent Fuel: mobile devices.

NDA method	Physical parameter assessed	Example of devices	Assay Duration (min)	Calibration interpretation	r (%)	s (%)
Gross Gamma Assay	Cooling Time	FORK [96,97]	1	Correlation law	5	20
Gross Gamma Assay	Cooling Time	PYTHON [90]	10	Correlation law	1	20
RTGS (NaI or CZT)	Presence of irradiated material	SFAT Safeguards [103]	10	Fission Product	Attribute test	
RTGS (CZT)	Presence of radioactive , material Burnup	SFAT Safeguards [103]	5	Fission Product, ^{137}Cs correlat. law	6-17	23-36
HRGS (HPGe)	LEU Burnup	SFM Safety Criticality [92]	10	$^{134}Cs/^{137}Cs$ or ^{137}Cs $^{106}Ru.^{137}Cs/^{134}Cs$ Correlation law	1	4
RTGS (CZT) + Passive neutron	MOX to LEU distinction	SMOPY Safeguards [100]	15	Comparison of NE/($^{134}Cs/^{137}Cs$) ratios	Attribute test	

Table XXVI: Performances of gamma and (gamma + neutron) NDA for Spent Fuel: fixed devices

NDA method	Physical parameter assessed	Example of devices	Assay Duration (min)	Calibration interpretation	r (%)	s (%)
HRGS	LEU BU, CT	BU PIT (Cogema La Hague) [90]	10	Isotopic Ratio $^{134}Cs/^{137}Cs$	1 -3	2-10
HRGS	LEU BU,CT	BNFL Thorp	10	$^{134}Cs/^{137}Cs$ or $^{134}Cs/^{154}Eu$, $^{106}Ru.^{137}Cs/^{134}Cs$ Correlation law	1	4
HRGS + passive neutron	LEU Average BU Extremity BU Pu mass	BU PIT (Cogema La Hague) [91]	10	Calculated Correlation Curve	0.1	2-5
Passive neutron	RBMK LEU BU	BUCK Safeguards [99]	10	Calculated Correlation Curve	1-10	3-5
Passive neutron	LWR, FBR Irradiation attribute test	CONSULHA	<1	qualitative	Attribute test	

methods are less data dependent, but unfortunately require heavy equipment [90].

Table XXVI presents actual information on instruments used at reprocessing facilities and reactors. One can easily see that estimated uncertainties are lower than the correspondent figures for mobile devices.

6 .CONCLUSIONS

The first evaluation of ESARDA NDA performance values was published in 1993. Almost 10 years later the Working Group for Standards and Non Destructive Assay Techniques (WGNDA) decided to review those values, to report about improvements and to issue new performance values. By taking into account the latest developments in NDA measurement technology, the present compilation also provides performance values for techniques that were not considered in the previous (1993) edition. Examples are:

- Gamma spectrometry with CdZnTe detectors, which are gaining more and more importance in Safeguards verification activities;
- the Gamma/X/Weighing (GXW) method for the simultaneous determination of uranium and plutonium isotopic ratios and element content in U and Pu bearing materials, showing potentials for a significant advancement for Safeguards verification measurements;
- Neutron multiplicity counting techniques, which help to significantly improve the measurement situation for poorly defined materials such as scrap, waste, and “dirty” (high-alpha,n) materials.

For a number of techniques already considered in 1993 the performance values have been essentially confirmed, but also changes both towards lower and higher performances have been observed in some cases. For the major neutron techniques (active and passive) the updated performance values conform more or less to the previous values, with one important exception: for small sample measurements using the inventory sample counter (INVS), notable improvements have been achieved as a result of systematic studies and improved calibration procedures.

For isotope abundance measurements by gamma spectrometry, the latest international round robin exercises practically confirmed the performance values for the ^{235}U enrichment determination, but also showed somewhat lower performances for plutonium isotope ratio measurements, compared to the 1993 values, which had been mainly elaborated on the basis of dedicated laboratory measurements. These results do not actually reflect a downgrade in the performances of the technique, but derive from a different perception of uncertainties and the feedback of a wider amount of experience. The results also underline the importance of sound and wide round robin exercises in assessing realistic pictures of performances of methods and techniques.

For U and Pu assay in solutions, the assessment confirmed (with some improvement) the very high performances of K-edge densitometry and XRF.

NDA performance values for waste measurements, not covered by the present document, are under further consideration by the WGNDA. The Group will also continue monitoring the performances of NDA techniques in use for Safeguards purposes, as well as of new emerging methods.

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Table XXVII: List of the ESARDA NDA Working Group members and observers

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Site definition in the Additional Protocol: How to focus on the real issues

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1. Abstract

Nuclear sites represent a key element in the Additional Protocol. In order to ensure that a state does not use collocated declared activities and buildings to cover up clandestine activities, the state is requested to give information about every building on the site and the IAEA is given far reaching access rights to all places on these sites. This seemed to be a logical approach that would not lead to many implementation difficulties.

The reality however showed to be slightly different. The definition of the site under the Article 2a(iii) declaration turned out to be one of the most important practical issues when preparing for the implementation of the Additional Protocol. Straightforward for standard facilities, it needs careful consideration for sites of a research centres or nuclear institutes in the middle of a university campus.

The site has to be carefully defined as it has a direct influence on the amount of information to be provided to the IAEA, the Agency's rights for Complementary Access, and the building owner obligations to grant this access in often a very short notice. Sites should be large enough to meet the objectives of the Additional Protocol. On the other hand there is no reason that buildings that do not contribute to the nuclear mission of the site should be part of it. The site boundaries should, therefore, be set in such way that no unnecessary burden is put on building operators that have no functional relationship with nearby nuclear activities.

2. Introduction

The site plays a central role in the expanded declaration under the Additional Protocol, as it requires the provision of information far beyond the information given in the Design Information. The site boundary definition in its turn has a direct influence on the Agency's far reaching Complementary Access right, which allow access to every building on a site and in principle every location inside a building with often very short notice of only 2h. This puts an up to now unknown obligation on building owners.

Sites have, therefore, carefully to be designed. They should be large enough to allow the IAEA to draw the conclusions of the absence of undeclared nuclear material or activities. But in line with the spirit of the Additional Protocol, there is no reason to include in the site buildings which do not contribute to the nuclear mission of the site. To monitor such buildings on a random basis assuming that every building on a site could in principle contribute to the nuclear mission would only absorb unnecessarily IAEA resources without added value for the IAEA.

Whereas the description of a site constituted by an active facility is in most cases straight forward, site of former nuclear research centres, technology parks, closed-down facilities or institutes on University campus may be more difficult to define. The paper therefore concentrates on those site issues, typical for the EU, where the size or the existence of the site is not a-priori obvious. It hereby develops a harmonised approach for all sites within the EU and gives guidance for the site definition.

3. European Union specific Issues

From the first discussions on site boundaries in the European Union, it became clear, that the interpretation of the Additional Protocol called for a detailed investigation for concrete cases in the

EU. The EU has a large number of closed-down facilities or Location-out-side facilities (LOF), which were no longer visited under traditional Safeguards, even though under the "strengthening measures" approved by the IAEA Board of Governors in 1992, Agency policy has been to continue to do design information verification on "closed down" installations until they have been verified as been "decommissioned" for Safeguards purposes. These installations may become subject again to reporting under the Additional Protocol under certain circumstances.

This issue is of particular importance for the research centres and technology parks in the EU. Where originally many of them constituted purely nuclear centres, over the years they have undergone major transformations and have diversified their activities so that nuclear activities and research constitute often less than 10% of their present activities. Most of the buildings, also the ones that were used for nuclear activities in the early days, are therefore nowadays used for non-nuclear activities, a typical situation as well for university institutes. To integrate these buildings in the site would on one hand create for the users and owners an unjustifiable burden and on the other hand direct the IAEA activities to areas of no-interest for the implementation of the Additional Protocol.

The NNWS in the EU comprise about 330 of such closed-down installations or locations, where about 200 of them are locations, which had only small or even very small quantities of nuclear material for non-nuclear use.

On the other hand, the EU has also a large number of locations holding only small amounts of nuclear material, 'Location outside facilities' (LOFs) or CAM holders ('Catch all MBA'). They are often no longer subject to classical Safeguards measures, but may become subject again to reporting under the Additional Protocol.

There are about 320 locations in the NNWS of the EU with less than 1eff kg of nuclear material and about 200 of them are locations with very small quantities of nuclear material for non-nuclear use, the so-called CAM holders.

In summary:

	NN-LOF	CAM	MBA/N-LOF	sum
active	81	195	173/46	495
closed-down/ decommissioned	85	125	91/32	333

4. Site declaration under Article 2a(iii)

4.1. Definition

(1) Article 18b of the Additional Protocol defines the elements that constitute a site: "Site means that area …..in the relevant design information for a *facility, including closed-down facility*, and *….location outside facilities where nuclear material is customarily used, including closed-down location outside facilities* where nuclear material was customarily used (this is limited to locations with hot cells or where activities related to conversion, enrichment, fuel fabrication or reprocessing were carried out). It shall also include all installations co-located with the facility or location, for the provision or use of essential services, including: hot cells for processing irradiated materials not containing nuclear material; installation for the treatment, storage and disposal of waste; and buildings associated with specified items identified …..under Article 2a(iv)."

(2) It follows from this definition, that the core of a site is always a facility or a LOF. Several facilities or LOFs can share one site, but a site cannot exist only without a facility or a LOF.

(3) Although Article 2a(iii)[1] requests the description of every building on a site, the Guidelines for reporting (see footnote 4) allow a site to be constituted of a single room. The IAEA recommends, that sites for facilities and nuclear LOFs should not be smaller than a single building, whereas sites for non-nuclear LOFs may be smaller than a single building.

4.2. Co-located buildings

(1) The site definition states that the site shall also include all installations co-located with the facility or location, for the provision or use of essential services, including: hot cells for processing irradiated materials not containing nuclear material; installation for the treatment, storage and disposal of waste; and buildings associated with specified items identified by the State under Article 2a(iv). But the Additional Protocol does not limit the co-located buildings to those given above.

(2) The interpretation of 'co-located …for the provision of essential services' has always been subject to discussions. However a pragmatic and reasonable approach could be to include within the site boundary the buildings in the neighbourhood of a nuclear installation or LOF, which are functionally related to the nuclear mission of this site. As a consequence 'essential services' can be understood – see footnote 2 – as hot cells, waste treatment, storage and disposal installations, buildings engaged in Annex I activities, that are needed for the nuclear mission of the site. These are buildings which in principle could cover up clandestine activities.

(3) Other services such as utility services, engineering and scientific support services, Computer services as well as some administrative and personnel-related services, like training, should be looked at on a case by case basis. They should be included if they exclusively serve the nuclear core on the site.

(4) The ownership would not be a criterion for exclusion or inclusion of a building housing essential services. Therefore different operators, owners, companies may exist on one site under Article 2a(iii). It is clear that this may cause legal and logistical problems in informing all concerned about ongoing inspections, in granting complementary access within the times foreseen, and in assembling the site declaration for the different buildings. In order to facilitate the implementation of the Additional Protocol, and based on a proposal of Member States, a 'site representative' will be designated by the State for each site as provided for in the draft new Euratom Regulation COM(2002)99 . This person would be responsible for the collection of the descriptions of the different buildings, the transmittal to the ESO, the information of the different operators of ongoing inspections and the facilitating of complementary access to all buildings on the site.

(5) The question what is meant by a 'building' should be answered pragmatically. A 'one-word' description (e.g. 'parking shelter') could be sufficient. Underground buildings, should be declared like all other buildings indicating the surface, number of floors etc.

4.3. Site boundary

(6) It may not always be obvious for the IAEA to understand why a certain building is not included in the formal site boundary when it is located in the vicinity of an installation or LOF. It may therefore be good practice to supplement the 2a(iii) declaration with supporting documentation explaining the functions of those buildings and the reasons why they should not be included in the formal site definition as is foreseen in the IAEA 'Guidelines' for reporting under II.12.

(7) Even if it is desirable that a site is constituted of one coherent area, the functional relationship between the buildings may require that a site would constitute of two disconnected islands. Also in this case it would be good practice to supplement such a declaration by a document detailing the function of the buildings between the islands and explaining the reasons why those buildings in the vicinity of a facility are not included in the site.

(8) Such an approach balances the wish of the State to limit the sites to the nuclear relevant buildings for logistical reasons with the need of the IAEA for transparency. Besides, the IAEA

[1] Article 2a(iii) "A general description of each building on a site, including its use and, if not apparent from this description, its contents…."

[2] Presentation of Davies Hurt, IAEA, on 'Site Definition' on the occasion of the ESARDA Working Group on the Additional Protocol/Integrated Safeguards in Helsinki, September 2001

[3] NN-LOFs containing nuclear material that has not been exempted

would always have the possibility of verifying adjacent buildings by a request under 2b(ii). This willingness for transparency could also be underlined by an invitation to the IAEA to schedule an announced visit to the buildings in the vicinity of a site prior to the Article 2a(iii) site declaration.

(9) Buildings declared in the BTC (Basic Technical Characteristics) as part of an MBA, i.e. which houses or housed a key measurement point, even if they contain no longer nuclear material, will be automatically part of the site. Especially research centres often have kept at least one such MBA, which contains or is licensed to contain small quantities of nuclear material spread over the centre. Concentrating these materials in one place and modifying the BTC accordingly may simplify the site definition significantly. An inconsistency between the site declaration and the BTC would automatically trigger at least requests for clarification and amplification under Article 2c.

(10) It should be underlined that the existence of a fence does not automatically determine the site boundary.

(11) Areas in the site definition that need a more detailed investigation are:

– Closed-down facilities or closed-down nuclear LOFs
– Active non-nuclear LOFs, including CAM holders (Catch all MBA)

5. Closed-down Facilities or LOFs

5.1. Definitions

(1) 4 categories of installations are currently differentiated:

– active installations;
– shutdown installations, which still have nuclear material;
– closed-down installations; and
– decommissioned installations.

(2) For the purpose of the Additional Protocol shutdown installations are treated the same way as active installations.

(3) Article 18c of the Additional Protocol: ”Decommissioned facilities or decommissioned location outside facilities means an installation or locations at which residual structures and equipment essential for its use have been removed or rendered inoperable so that it is not used to store and can no longer be used to handle, process or utilise nuclear material.”

(4) Article 18d of the Additional Protocol: “Closed-down facility or closed-down location outside facilities means an installation or location where operations have been stopped and the nuclear material removed but which has not been decommissioned.”

5.2. Relevance for the Additional Protocol

(1) It follows from the definition of the site as given in 3.1, that closed-down installations (facilities or LOFs) constitute a site.

(2) Many of these installations with the label 'closed-down' have been largely dismantled over the years: the structures and components essential for the use within the nuclear fuel cycle have been completely removed and very often the buildings have been made available for other (non-nuclear) activities.

(3) In the past traditional Safeguards measures came in practice to an end when the safeguards authorities had verified that the nuclear material was definitively removed from the installation. The installation was then labelled as closed down. However, the verification of closed down

[4] “Guidelines and Format for Preparation and Submission of Declarations Pursuant to Article 2&3 of the Model Protocol Additional to the Safeguards Agreements”, August 1997

[5] Current IAEA policy clarifies that decommissioned has to be understood as decommissioned for Safeguards purposes. Dismantling activities may still continue after the location has received the status ‘decommissioned for Safeguards purposes’.

installations, or installations in the process of being decommissioned, is an integral part of the IAEA strengthening measures under existing legal authority (Part 1). The Safeguards authorities therefore need to collect information about the present status of such installations.

(4) Once an installation is confirmed to be decommissioned for Safeguards purposes, it ceases to be the core of a site.

5.3. Closed-down LOFs

(1) As detailed in the site definition of Article 18b a closed-down LOF still constitutes a site if it contains a hot cell or if it was involved in activities related to conversion, enrichment, fuel fabrication or reprocessing. LOFs involved in such activities belong to the family of nuclear-LOFs (see 6.1(2)).

(2) It is clear from the above site definition that closed down non-nuclear LOFs (NN-LOF) do not constitute a site in itself. (It is however not excluded that they may be part of a site that is formed around another installation.) Therefore, closed-down NN-LOFs in the EU not included in other sites will not be subject to Article 2a(iii) reporting. A closed-down NN-LOF can be regarded as decommissioned, if it does not include a hot cell.

(3) Often the nuclear material in a LOF was kept in a store, safe or a laboratory (without a hot cell). These places very often do not have ‘essential equipment’. In this case the removal of the nuclear material and therefore the status closed-down would at the same time mean, that this LOF has the status ‘decommissioned’. The judgement has to be made on a case by case basis depending whether it has specific nuclear material related equipment.

5.4. Transfer from closed-down to decommissioned

(1) Euratom Safeguards is presently contacting the operators of all installations in the European Union labelled 'closed down' in order to establish their precise decommissioning status. The inspectorates will visit former nuclear installations that claim to have reached the status “decommissioned” to confirm the status.

(2) Once the decommissioned status is confirmed these installations or LOFs do not constitute a site anymore.

6. Active Location outside facilities (LOF) and CAM holders

6.1. Introduction

(1) A Location outside facility (LOF) means any installation or location, which is not a facility, where nuclear material is customarily used in amounts of 1 effective kg or less (INFCIRC 540, Article 18j).

(2) Depending from the use of the nuclear material LOFs are divided in 2 categories: nuclear LOFs and non-nuclear LOFs. Nuclear material in non-nuclear LOFs (NN-LOFs) is used for purposes, which are not nuclear fuel cycle related. There are actually 46 such N-LOFs and 81 NN-LOFs in the NNWS of the EU (see table above).

(3) The catch-all-MBA (CAM) (code WWWW) in the EU is a virtual NN-LOF including a large number of CAM-holders, currently 195, spread over Europe holding each very small quantities of nuclear material. The sum of the quantities of the nuclear material in all CAM-holders in the EU is less than 1 effective kg.

(4) The above would mean that under the present arrangements some 320 locations with small or even very small quantities of nuclear material in the EU would each constitute a site with map and detailed description of buildings on that site. This seems to be an unnecessary burden for the IAEA, the operator and the State or Regional Authority without a corresponding added value to the IAEA, if it concerns places with no functional relation to the nuclear mission.

(5) For the further discussion, on how nuclear material in LOFs can be treated, some definitions

may be recalled to support the decision process:

6.2. Definitions

(1) **Nuclear activities**: Nuclear activities are activities that are related to the nuclear fuel cycle[6], such as defined in Article 18a of the Additional Protocol[7].
(2) **Non-nuclear activities** therefore would be all other activities not covered by (1).
(3) **Recoverable**: nuclear material is considered recoverable unless Euratom and the Agency have agreed that it is practically irrecoverable[8]. Recoverable nuclear material may qualify for exemption from IAEA Safeguards, but not for termination.
(4) **Irrecoverable**: nuclear material, on which Euratom and the Agency agreed that it is not recoverable. Such material qualifies for termination of IAEA Safeguards.
(5) **End-use form**: final use of the nuclear material, e.g. shielding containers. So: "DU ingots acquired by a LOF for making shielding containers is not in the end-use form, but the shielded containers are[8]." The same would apply for aircraft counter weights.
(6) **Exemption**: nuclear material, which could qualify for exemption based on use (Article 36b of the Safeguards Agreement) is natural uranium, depleted uranium and thorium. Although the Safeguards Agreement does not preclude the exemption of plutonium, high or low enriched uranium, current IAEA policy8 recommends that nuclear material in such forms should normally not be exempted for non-nuclear use.
(7) Exemption of nuclear material from IAEA Safeguards based on quantities (Article 37 of the Safeguards Agreement) is normally not used in the EU.

6.3. Possibility of Exemption under INFCIRC 193:

(1) The Safeguards Agreement foresees the possibility of exempting nuclear material from IAEA Safeguards or terminating IAEA Safeguards on nuclear material under certain circumstances:
(2) Article 36, Article 13 and Article 38 of the Safeguards Agreement stipulate:
Article 36:"At the request of the Community, the Agency shall exempt Nuclear material from Safeguards under this agreement, as follows:
(a) special fissionable material, when it is used in gram quantities or less as a sensing component in instruments;
(b) Nuclear material when it is used in non-nuclear activities in accordance with Article 13, if such nuclear material is recoverable;
(c) plutonium with an isotopic concentration of plutonium-238 exceeding 80%."
Article 13: "Where nuclear material subject to Safeguards under this Agreement is to be used in non-nuclear activities, such as the production of alloys or ceramics, the Community shall agree with the Agency, before the material is so used, on the circumstances under which the Safeguards under this Agreement on such material may be terminated."
Article 38: if exempted nuclear material is to be processed or stored together with safeguarded nuclear material, provisions should be made for the reapplication of Safeguards thereto."
(3) From the above follows, that for material - such as D, N or Th - to qualify for exemption it has to be used in non-nuclear activities. As per definition these places have no nuclear fuel cycle related activities, their relevance for IAEA safeguards is negligible.

[6] "Nuclear fuel cycle - a system of nuclear installations and activities interconnected by streams of nuclear material. The characteristics of the fuel cycle may vary widely from State to State, from a single reactor supplied from abroad with fuel, to a fully developed system. Such a system may consist of uranium mines and concentration (ore processing) plants, thorium concentration plants, conversion plants, enrichment (isotope separation) plants, fuel fabrication plants, reactors, spent fuel reprocessing plants and associated storage installations. The fuel cycle can be 'closed' in various ways, for example by the recycling of enriched uranium and plutonium trough thermal reactors (thermal recycle), by the re-enrichment of the uranium recovered as a result of spent fuel reprocessing or by the use of plutonium in a fast breeder reactor. ", IAEA Safeguards Glossary, 2001 edition, p.37

[7] In accordance with Article 18a of the Additional Protocol 'nuclear fuel cycle related' means related to conversion or enrichment of nuclear material, nuclear fuel fabrication, reactors, critical facilities, reprocessing of nuclear fuel and processing of intermediate or high-level waste containing plutonium, high enriched uranium or uranium-233.

[8] IAEA note on the "Treatment of Non-nuclear LOFs under the Additional Protocol dated 10/6/02

(4) The exemption remains valid as long as the nuclear material is not permanently transferred to another location. In this case it has to be reported to the Safeguards authorities.
(5) Furthermore, Safeguards could be terminated on nuclear material in NN-LOFs and CAM holders when it is used in non-nuclear activities and the material is irrecoverable.

6.4. Relevance of 'Exemption' Status under the Additional Protocol

(1) Sub-Articles 2a(iii) or 2a(vii) of the Additional Protocol may be relevant to exempted material.
(2) Places holding only nuclear material on which safeguards has been terminated in accordance with Article 13 of the safeguards agreement would no longer constitute the core of a site. So no reporting would be necessary under Article 2a(iii) of the Additional Protocol, nor under Article 2a(vii).
(3) Provided that the location and quantity of exempted nuclear material is known, there is no proliferation gain in receiving further information about the sites of non-nuclear LOFs which contain no nuclear fuel cycle infrastructure. It is therefore proposed that such locations are not reported as sites under Article 2a(iii).
(4) LOFs holding exempted nuclear material will never have to report under Article 2a(vii)-b of the Additional Protocol, because the limits for reporting under Article 2a(vii)-b correspond to 1 kg_{eff}. LOFs per definition never hold more than 1 kg_{eff}.

7. Treatment of installations holding more than 1 kg_{eff} that qualify for exemption

(1) The criteria explained in para 5.3 for exempting nuclear material and its consequence under the Additional Protocol as explained in para 5.4 apply as well for installations holding more than 1 kgeff .
(2) Installations with exempted material exceeding 1 eff kg may be subject to reporting under Article 2a(vii)-b of the Additional Protocol:
- Article 2a(vii)-b calls for: "Information regarding the quantities …and uses at each location of Nuclear material exempted from Safeguards pursuant to paragraph 36(b) of the Safeguards Agreement but not yet in a non-nuclear end-use form, in quantities exceeding those set out in paragraph 37 of the Safeguards Agreement…"
(3) Exempted material in such installations therefore has to be reported under Article 2a(vii)-b when it is not in its end-use form AND it is in quantities exceeding those of Article 37[9]: "…
(b) 10 tonnes in total of natural Uranium and depleted Uranium with an enrichment above 0,5%
(c) 20 tonnes in total of depleted Uranium and depleted Uranium with an enrichment of 0,5% or below
(d) 20 tonnes of Thorium…"
(4) It is expected that very few locations in the EU holding nuclear material, which qualify for exemption, will be subject to reporting under Article 2a(vii).

8. Conclusion

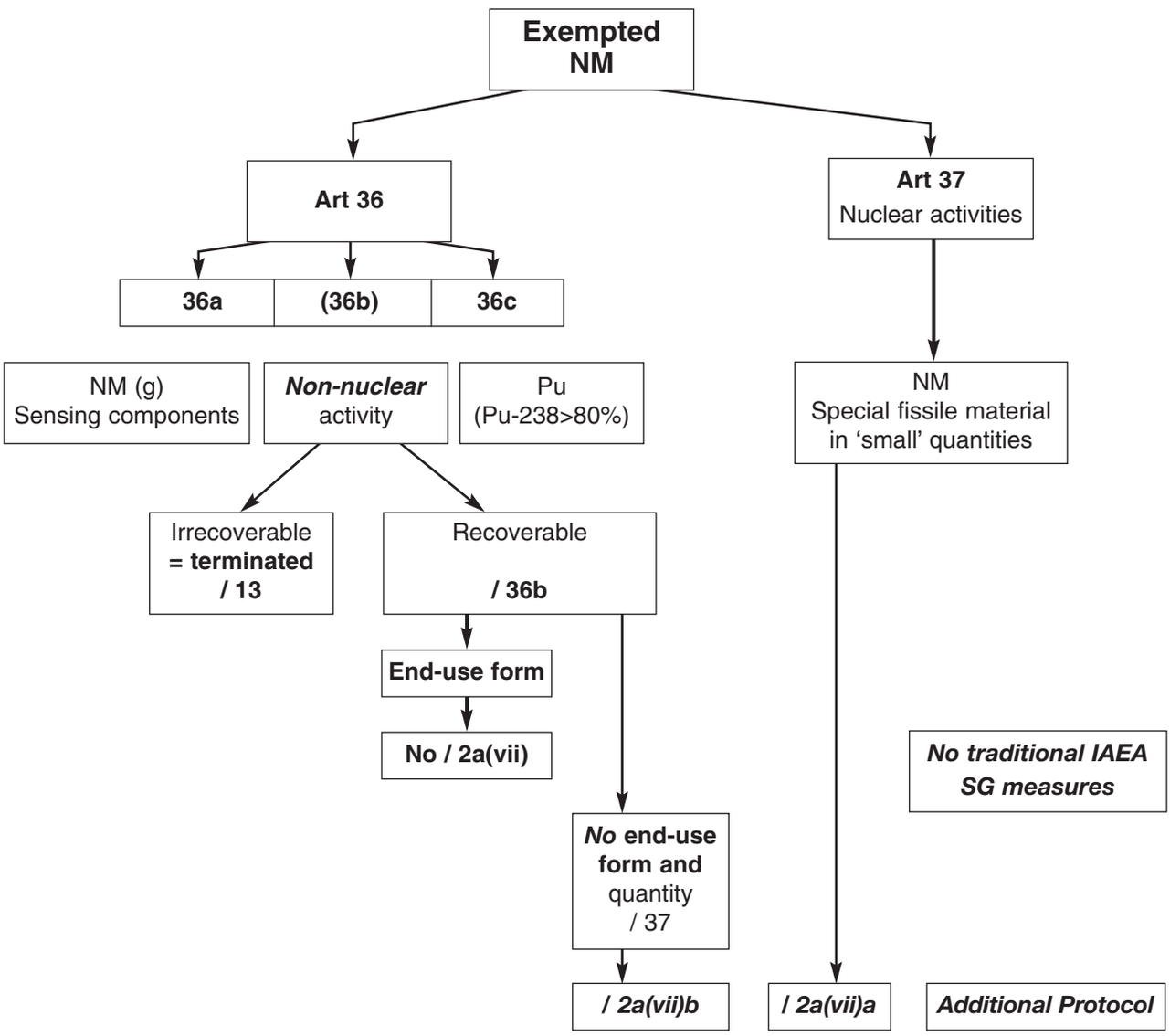
(1) Once the decommissioned status is confirmed these installations or N-LOFs do not constitute a site anymore.
(2) Closed-down NN-LOFs (including CAM holders) do not constitute a site.
(3) LOFs and other places holding only nuclear material exempted from IAEA Safeguards need not be reported as sites under Article 2a(iii).
(4) Nuclear material on which Safeguards is terminated is not subject to reporting under Article 2a(vii).

[9] The IAEA clarified in its Note referenced in Footnote 8 that the Article 37 quantity limits are meant per location.

(5) Nuclear material exempted from IAEA Safeguards under Article 36(b) is subject to reporting under Article 2a(vii)(b) if it is

– Not yet in non-nuclear end-use form and
– Exceeds for the location concerned the quantities identified in Article 37 of the Safeguards Agreement

9. Summary of Exemption issue with respect to the Additional Protocol :



Aspects of Unannounced Inspections - A View of the ESARDA WG on Integrated Safeguards

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Abstract

Unannounced inspections (UI) are already provided for in the INFCIRC/153 type Safeguards Agreements. In proposed approaches for the application of Integrated safeguards UI also play a major role. They will allow the Agency much faster access to relevant locations in a facility than any other type of inspection. Additional valu-able synergy effects for the Agency's safeguards measures under the Additional Protocol can be expected from the combination of UI with Complementary Access (CA).

Application of UI has advantages and disadvantages. On the one hand it places the potential diverter in a per-manent state of uncertainty and can be used to detect and deter from undeclared activities in a facility and, thus, can be an efficient and cost effective tool to cover a range of diversion scenarios. However, on the other hand, UI imposes additional burden to facility operators and increases the danger that inspections may interfere with planned operational activities.

This paper discusses the pros and cons of the application of unannounced inspections from the view of the facil-ity operators and national authorities as discussed within the ESARDA Working Group on Integrated Safe-guards .

Keywords: Additional Protocol; Unannounced Inspection; Integrated Safeguards

1. Introduction

The ESARDA Working Group on Integrated Safeguards was created in 2000 to discuss aspects of Integrated Safeguards from the view of the operators of nuclear facilities in the European Union. This paper reflects the current status of discussions within the group on the use of unannounced inspections (UI) in Integrated Safe-guards approaches for nuclear facilities.

In integrated safeguards approaches for specific facility types, randomly scheduled unannounced inspections are considered as one possible and cost-effective measure in the inspection scheme. Unannounced inspections are already provided for in Paragraph 84 of INFCIRC/153 (corr.):

However, the Agreement should also provide that, as a supplementary measure, the Agency may carry out without advance notification a portion of the routine inspections pursuant to paragraph 80 above in accordance with the principle of random sampling. In performing any unannounced inspections, the Agency shall fully take into account any operational programme provided by the State pursuant to paragraph 64(b). Moreover, whenever practicable, and on the basis of the opera-tional pro-

[1] General disclaimer : The views expressed in this paper are the views of the individual members participating in the ESARDA Integrated Safeguards Working Group, and do necessarily represent the official view of their organisation. The personal view of some participating members may differ of the opinion expressed in this WG paper on some aspects.

gramme, it shall advise the State periodically of its general programme of announced and unannounced inspections, specifying the general periods when inspections are foreseen. In carry-ing out any unannounced inspections, the Agency shall make every effort to minimize any practical difficulties for facility operators and the State, bearing in mind the relevant provisions of paragraphs 44 above and 89 below. Similarly the State shall make every effort to facilitate the task of the inspectors.

Although UI are no new tool in the set of possible safeguards measures, use of it is made at present only in a few approaches like in the LFUA approach for gas centrifuge enrichment plants. Now, UI are proposed for routine use in Integrated Safeguards approaches for differ-ent types of facilities.

2. Example of successful application of UI: LFUA inspections

As an outcome of the Hexapartite Safeguards Project, the concept of 'Limited Frequency Unannounced Access' (LFUA) to cascade halls was developed for gas centrifuge enrichment plants which is applied now since many years in the URENCO enrichment plants in the EU.

As a deterrence measure to undeclared HEU production, the LFUA safeguards concept establishes an inspec-tion regime which allows access to the cascade halls 24 hours a day and 365 days a year with only 2 hours notification. The purpose and the manner of execution of these LFUA inspections are well defined. The purpose of these inspection is the re-verification of the routing of the main cascade header pipework and their intercon-nections, the confirmation that no additional feed and take-off points (blind flange connections) have been intro-duced and no undeclared feed or take off stations are present. This is achieved through the visual observation of the relevant cascade structures and, as agreed, NDA measurements to confirm the absence of HEU. Some design documentation, e.g. schematics and photos are provided by the operator and verified during an initial design information verification visit, these form the basis of the re-verification. The LFUA visit starts with an initial meeting to collect the relevant documents, to identify any changes and to carry out other preparatory work, and it continues with a continuous walk through the cascade halls on an agreed route, with special arrange-ments to undertake agreed NDA activities.

The concept and the execution of LFUA is judged by the operators of the facilities concerned as an effective and successful safeguards measure. They attribute this efficiency and success essentially to the facts that firstly these inspections have clearly defined objectives, namely to confirm that there are no changes in the configura-tion of the pipework of the cascades, etc., secondly, because the plants have large parts which are repetitive in nature and do not change after commissioning, and thirdly there is a clearly defined simple and regular manner of execution. Whenever these inspections take place, this grants for both sides a well defined and steady situa-tion where mutual duties and rights are well known. It also allows the operator to instruct a sufficient number of staff members and thereby guarantee at any time the availability of a trained escort to the inspectors on their walk through the cascade halls.

3. Different Aspects of Applying UI

Unannounced inspections will allow the Agency much faster access to relevant locations in a facility than any other type of inspection. The advance notice time for ad hoc inspections and routine inspections in most cases amounts to at least one week, for transfers into or out of the state and for facilities other than reactors with inventories of plutonium or uranium enriched to more than 5% it amounts to at least 24 hours (see Article 83 of INFCIRC/153). In the case of an unannounced inspection this advance warning time is reduced to the time period between showing up at the gate and having access to the desired strategic point within the facility.

Additional valuable synergy effects for the Agency's safeguards measures under the Additional Protocol can be expected from the combination of UI with Complementary Access (CA). Only when conducted in conjunction with a design information verification visit or an inspection, CA can be performed with an advance notice of at least 2 hours. In all other cases the advance notice time has to be at least 24 hours. Complementary access in conjunction with unannounced inspections will allow for the fasted

access to any place on a site and will consti-tute the greatest effect of surprise concerning the operator.

From the Agency's point of view, the unpredictability aspect of UI should be very appealing, as long as truly unannounced inspections can be carried out. UI places the potential diverter in a permanent state of uncertainty and can be used to detect and deter from undeclared activities in a facility and, thus, can be an efficient and cost effective tool to cover a range of diversion scenarios.

From the operator's point of view, it is his main task to run the plant in a safe and cost-effective manner. His planning and management practice aim to reduce all elements of unpredictability and surprise as far as possible to meet his operational goals. If a state and the operator are ready to accept UI, this first of all underlines their strong commitment for the non-proliferation objectives of international safeguards.

Additional burden for the operator may stem from requirements to regularly inform the Agency in advance of certain activities or events the Agency regards as safeguards relevant (e. g. temporary access restrictions due to contamination, handling of large containers in the reactor hall, repair work on fuel elements). It has to be kept in mind that some of those events can only be communicated after their occurrence and others only on a short term basis to maintain the necessary operational flexibility.

The desired effect, from a safeguards point of view; of unpredictability of unannounced inspections increases the danger that UI may interfere with planned operational activities or that intended inspection activities may be delayed or even not possible to be performed due to the lack of available operator resources. Such situations should (or must) not be interpreted as an unwillingness of the operator to cooperate and thus lead to a failure to meet the inspection goals. An other fact to be kept in mind is that UI is for most states and types of facilities a new element for which no experiences yet exist. We don't believe that a mechanistic schema of performance evaluation, as it is applied to current inspections, can contribute to the success UI.

In order to be able to make a decision whether they can in principle accept and support UI, the states and the operators need clear and transparent information from the Agency about the conditions and consequences of this decision. Vital points are for instance:

- the clear specification of the objectives of unannounced inspections for different types of facilities;
- the clear specification of requirements and measures to meet the objectives taking into account the different practicalities and operating conditions;
- the criteria/procedure of the Agency to measure the goal attainment of UI in the case of difficulties to perform all planned activities as expected;
- conditions/events that are regarded by the Agency as anomalies or unsatisfying results;
- the follow up actions and procedures in cases ofanomalies or unsatisfying results.

UI also poses high requirements concerning the skills of the inspectors in charge of the inspection. They will be confronted with all types of plant circumstances and must be able to judge the situation on the spot based on a sound expert knowledge. Besides the technical knowledge they have to be acquainted with the plant manage-ment procedures and with legal constrains, e. g. the operating licence requirements.

4. General implementation considerations

The ESARDA WG IS welcomes the intention to use UI in Integrated Safeguards approaches. We believe that within the EU states the logistical conditions (travel, visa, etc.) to implement UI and fully maintain the element of surprise can be met. We also believe that other issues necessary for the implementation can be solved.

Concerning the travel logistics, we don't see great difficulties for the Agency to carry out truly unannounced inspections within the member states of the European Union. Inspectors can travel from the IAEA headquarter in Vienna without passing border controls in the EU states acceded to the "Schengen" agreement.

To allow states and operators to make their decision to fully support the implementation and to prepare for the necessary arrangements, clear and transparent information on conditions and conse-

quences of UI compared to alternative measures should be provided by the Agency. This means in particular the assessment of the UI goal attainment by the Agency and the follow up procedures in cases where goals are not sufficiently attained.

The ESARDA Working Group on IS is well aware of the fact that the successful implementation of UI will require a high level of co-operation from all sides, i.e., the Agency, the State/RSAC and the operator. The plant opera-tors are willing to do their best to contribute to that task.

For a successful implementation, the objectives of UI for the different types of facilities and the measures to meet these objectives should be exactly defined and be transparent to the operator. From the side of the Agency, we expect that the objectives are set in such a way that they can be met even under difficult circum-stances. The measures required to meet the objectives must be flexible and robust enough to take into account a wide range of actual circumstances in the plant without automatically leading to a failure in case of non-optimal conditions. Inspections within the LFUA regime proved to be successful because they are well defined concern-ing their objectives and their course of actions. The frequency of UI is expected to be in most cases much less than the frequency of LFUA inspections. So the effect of experiences with past UI will be much less on both the operator's and the Agency's sides. All the more a well defined frame for the course of UI is needed with a clear definition of the mutual duties and rights. Only a well defined set of rules allows the operator to instruct his personnel in such a way that it can cope with truly unannounced inspections at any time. We believe that such a well defined set of rules is indispensable for a successful application of UI.

The intended purpose of a nuclear power plant is to produce energy under economic principles. Power plants in Europe are often highly automated and are normally run with a minimum of personnel. Nobody can expect that additional capable staff for safeguards purposes is on stand-by around the clock for an event that will happen on the average once in 5 years. This applies specifically to unannounced inspections outside normal working hours.

Paragraph 84 of INFCIRC/153 states in its last sentence "...*In carrying out any unannounced inspec-tions, the Agency shall make every effort to minimize any practical difficulties for facility operators and the State, bearing in mind the relevant provisions of paragraphs 44 above and 89 below. Similarly the State shall make every effort to facilitate the task of the inspectors."*

Paragraph 89 of INFCIRC/153 states " *... that the state shall have the right to have inspectors accompa-nied during their inspections by representatives of the state...*"

To take into account these requirements, the practical implementation of UI should be based on agreed detailed procedures specific for the individual facility in question. These procedures should specify all possible actions to be carried out during the UI at this facility with a range of time granted to the operator to prepare for the neces-sary supporting activities.

As a general framework, we propose the following rules for the implementation of unannounced inspections:

- Unannounced inspections should be allowed to be carried out at any time with no advanced notice to maintain the effect of unpredictability, this means, even outside normal working hours.
- The operator will grant access to strategic points in the facility within the shortest possible time for the pur-pose of visual observation and other agreed measures. These strategic points however will not be permitted to be within those areas defined as requiring managed access under the Additional Protocol.
- For all additional activities the Agency may have planned for this UI, it cannot expect that they will be carried out immediately. The operator shall be granted the necessary time to make his preparations and the state representative/ESO inspector the time to arrive at the facility.
- UI can also be performed as a short notice inspection. If the UI is performed with an advanced notice, the preparation time granted to the operator after arrival on site can be shortened accordingly.

5. Examples of issues to be covered in the procedures

- The operator is not allowed to grant access to persons whose identity and authorisation has not been veri-fied. What is the procedure to notify the operator of the event that an unan-

nounced inspection is taking place, how will the inspector identify himself to the operator, and how will he prove his authorisation for this inspection?

- What are the specific objects of an UI in this specific facility, and what are the measures to meet these objects?
- Are different types of UI foreseen for this facility, and how do they differ from each other in objectives, meas-ures, and notification time? Are the measures in line with the objectives of the UI?
- What is the information to be reported about the operational programme of the facility (which events, plans, activities, level of detail, time frame, etc.)? For what purpose and how will this information be used? (For a truly random UI it cannot influence the time of the inspection.)
- Which are the strategic points, where the inspector may wish to go, and what is the maximum acceptable time to get from the gate to these places, depending on conditions like during/outside working hours, during fuel receipts or shipments, during maintenance activities, etc.?
- What has to be done, if an inspector could not gain access within the specified time (e.g., the inspection is delayed because of health or safety requirements)?
- In case of unannounced inspections not all activities can be performed in the time normally spent for them when scheduled in advance. There may even be inspections by national safety authorities at the same time that consume additional manpower.
  For each supporting activity which an inspector may require from the operator, the range of time (working days) allowed for preparation should be specified, e. g.:
  - Updating of the book inventory on the basis of accounting and operating records may not be possible, if accounting and operating records are not available or not up-to-date at the facility.
  - Item counting and verification for gross defects of spent fuel may not always be possible due to the lack of personnel to operate the bridge, maintenance work on the bridge or other scheduled work using the bridge.
- What are the agreed measures to be applied to freeze the situation at a strategic point, if regarded neces-sary?
- What are possible follow up actions or substitutes, if planned activities cannot be performed due to opera-tional reasons?
  We should keep in mind that operators of nuclear facilities are trained to plan for the worst case, a case where everything that can go wrong, goes wrong. This view also is the main driving force to aim at clear and detailed procedures for the course of unannounced inspections. The desire of a set of well defined rules arises from this attitude and does in no way indicate a disapproval of unannounced inspections.

6. Conclusions

Application of UI has a clear advantage for the integrated safeguards approach. It places the potential diverter in a per-manent state of uncertainty and can be used to detect and deter from undeclared activities in a facility and, thus, can be an efficient and cost effective tool to cover a range of diversion scenarios. The operators and other involved organisation in EU countries recognise it and support the use of Unannounced Inspections.

However, the operator's main task is to run the plant in a safe and cost-effective manner. His plan-ning and management practice aim to reduce all elements of unpredictability and surprise as far as possible to meet his operational goals. In the light of the experience of LFUAs in centrifuge enrichment plant in Europe, it appears that a clear procedure established in advance is a pre-requisite to successful implementation.

For Unannounced Inspections to be a win-win situation for the IAEA and inspected organisations, we strongly feel that clear rules and procedures must be prepared from the beginning, adapted to the specificities of each plant, and covering practical modalities, scope of inspection and reasonable guidelines to assess that the UI goals have been achieved. We trust that all involved parties share this view.

Integrated safeguards: a case to go beyond the limits

Consequences of boundary limits set to the reduction of "classical safeguards measures" on efficiency and resources allocation in Integrated Safeguards

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Abstract

The ESARDA Integrated Safeguards Working Group has been created in 2000. It is a forum of exchange on Additional Protocol implementation (e.g. site definition, …) as well as on the development of Integrated Safeguards. It also analyses the Integrated Safeguards overall framework as it is developed by the IAEA and its consequences on operators, member states and safeguards authorities. Integrated Safeguards have reached a milestone with the achievement by the IAEA of its conceptual framework. Looking back at what was expected from Integrated Safeguards shows that some assumptions (one PIV / year, recoverability within existing timeliness goal, Unirradiated Direct Use material) applied to the current Integrated Safeguards framework put constraints on what can be achieved. Consequently it puts a strong limitation on savings expected from "classical" safeguards verification measures and it constrains the value expected from a greater use of qualitative tools. The paper discusses some of those boundary conditions, and suggests a better balance between quantitative and qualitative tools. The ultimate goal is to enhance the safeguards effectiveness through a flexible implementation of Integrated Safeguards.

1. ESARDA Integrated Safeguards Working Group: fostering exchange

The ESARDA Integrated Safeguards Working Group has been created in 2000 after the signature of the Additional Protocol in the European Union and when the IAEA was designing the Integrated Safeguards framework with the support of several Member States Support programme tasks. The goal of this working group is to evaluate the implementation consequence of the Additional Protocol and Integrated Safeguards, to foster an harmonised understanding between its members and to provide an input to the IAEA. By early 2003, the working group comprises some 20 members: national authorities and operators from 8 countries, Euratom Safeguards and one observer from Switzerland. The IAEA is also regularly participating at the meetings, and it proves a valuable way to increase mutual understanding.

A key output of the working group are the rich exchanges of views which characterise every meeting: feed back of experience and actual examples, sharing questions, discussing views and interpretation on selected topics, … In addition to the direct benefit to each participant, it allows for the emergence of an harmonised view on key issues related to the implementation of the Additional Protocol and the development of Integrated Safeguards.

The agenda covers a wide range of topics. As an illustration, recent work includes:

[1] General disclaimer: The views expressed in this paper are the individual views of the members participating in the ESARDA Integrated Safeguards Working Group, and do not necessarily represent the official view of their organisation. The personal view of some participating members may differ of the opinion expressed in this WG paper on some aspects.

Additional Protocol declaration, with a special emphasis on site definition and more work scheduled on R/D declaration

Practicalities on implementing proposed Integrated Safeguards approaches, with a specific focus on Unannounced Inspections for LWR

An overall analysis of the Integrated Safeguards conceptual framework, goals and approaches, which is further developed below

2. Integrated Safeguards: founding definitions

Strengthened Safeguards and more specifically the additional protocol have introduced a new major component in the overall safeguards system: the coverage of the undeclared path. The combined objective of in the implementation of a comprehensive safeguards agreement and additional protocol is to provide credible assurance of both the non-diversion of nuclear material from declared activities and of the absence of undeclared nuclear material and activities in the State as a whole. Once this assurance is obtained, the IAEA can apply integrated safeguards.

Integrated Safeguards is defined as the optimum combination of all safeguards measures available to the Agency under comprehensive safeguards agreement and additional protocol. It can be achieved by maximising the synergy or the complementarities of the measures while not reducing the overall strength or effectiveness of the safeguards system.

The verification effort on declared nuclear material under a comprehensive safeguards agreement alone is based on the assumption that undeclared nuclear activities necessary to produce nuclear-weapons-usable material, e.g. undeclared reprocessing or enrichment facilities, may exist undetected. Under a comprehensive safeguards agreement together with an additional protocol, the Agency's ability to draw a conclusion on the credible assurance of the absence of such undeclared activities in the State as a whole creates the potential for reductions in verification effort on declared nuclear materials that would need further processing to produce nuclear-weapons-usable material, e.g. irradiated fuel and depleted, natural or low enriched uranium. Diversion of nuclear material is an important element but one among other numerous elements and indications available to the Agency (thanks to AP measures) to detect a nuclear weapons acquisition program.

Now integrated safeguards is not simply about reducing so-called classical safeguards. It is to **find the optimum combination of all available measures** including nuclear material accounting, verification of non-diversion of nuclear material, complementary access to a wider range of locations, information on manufacturing, import and export of nuclear material but also key equipments or components, etc…

Integrated safeguards calls also for efficiency: once Integrated Safeguards is routinely implemented and the IAEA passed over the initial cost of evaluating the initial declaration, Integrated Safeguards are expected not to require any additional IAEA resources. In practical terms, the resources required for the implementation of the Additional Protocol (information treatment and evaluation, complementary access, reporting and drawing conclusions) should come from the savings in PDI's and equipment the IAEA makes in safeguarding nuclear facilities. Integrated safeguards is giving rise to high expectations: efficiency aiming at a "zero sum" game along with a strengthening of the safeguards regime. It may be too ambitious to expect a real "zero sum" game; but it shall also be recognised that even if the overall safeguards budget were to be increased, there will always be a financial limitation calling for efficiency and optimisation of resources allocation, as it is today the case for "classical" safeguards.

The apparent reduction in effectiveness through a decrease of the use of classical safeguards measures (obtained for instance through a different setting of parameters such as timeliness, detection probability, …) shall be more than compensated by an increase in the safeguards effectiveness through the use of the other measures. Calling for efficiency and effectiveness, integrated safeguards must be a "whole is more than the sum of the part" game. The underlying assumption is therefore that Additional Protocol measures will bring a higher return on investment, i.e. a higher effectiveness. Consequently, the allocation of resources between all measures available under integrated safeguards (AP measures and classical verification of non-diversion) should be such that those AP measures can be fully implemented. While recognising that it is difficult today to

measure the effectiveness of the Additional Protocol measures, we believe that the only way to get enough experience is to fully implement them.

When developing the Integrated Safeguards conceptual framework and the corresponding safeguards approaches, the IAEA used a series of general rules or boundary conditions that largely determined (limited) the scope for possible resource savings in the different facilities. Using such boundary conditions is certainly a valid initial step towards developing integrated safeguards. However, capping the savings constrains the freedom to find a real optimal resource allocation between all available measures. To go further into the development of Integrated Safeguards, those boundary limits shall be recognised and their need challenged. Those boundary limits are presented and discussed below.

3. Boundary Conditions used in the Development of Integrated Safeguards Approaches

3.1. The Need for One PIV per Year: can we meet SIR needs differently ?

"The requirement for an annual physical inventory verification (PIV) will continue to apply under integrated safeguards."

This requirement could seem justified by the present rule of performing a yearly material balance evaluation and report in the SIR, but it puts a de facto limitation to any revision of the timeliness goal beyond one year. It could be questioned whether, under Integrated Safeguards conditions, it is really necessary to perform such yearly material balance evaluation through a PIV only to back-up an annual reporting requirement.

It could be envisaged that every calendar year a random sample of similar facilities in a State is selected for PIV's, the results of which could then be extrapolated to the entire population of similar facilities in the States using standard statistical methods. Or in the case of LWR's an option could be to drop the "PIV equivalent" and to only perform the PIV when the core is open and when all assemblies are available for verification, even when this would mean that the period between subsequent PIV's would then be longer than 1 year for reactors with longer cycles.

For other types of facilities also, the need of an annual PIV could be re-assessed. In the case of spent fuel storage in difficult to access conditions, the use of advanced C/S technology and possibly some random inspection may give way to a lower frequency of PIV?

3.2. No Change in Safeguards Measures on Unirradiated Direct Use Material

The IAEA appears still to be reluctant to consider a reassessment of the verification requirements for Unirradiated Direct Use Material (UDU) apart from what has been decided for LWR with MOX fuel. These materials include separated Pu and fresh HEU. This requirement is justified by the fact that the IAEA physical model does not go beyond the production of UDU. However, between the acquisition of UDU and the final production of the nuclear weapon lays the entire weaponisation process, a process that certainly has a distinct probability of being detected by the measures of the Additional Protocol. By inclusion in the physical model and the acquisition paths of those weaponisation indicators that are available in the open literature, room may be created for also reducing safeguards measures on UDU and the facilities where these materials are created or processed.

The IAEA may also recognise that, for plutonium, the isotopic composition and physical form could be taken into account in defining the safeguards approach in other facilities than LWR with MOX. The actual isotopic composition of most of the plutonium being recycled, and the industrial maturity of recycling policies could become part of the evaluation. The intensity of verification could be adapted to the "strategic value" of the material. One could imagine an approach where these facilities are looked at on a case-by-case basis, taking into account the State fuel cycle specific characteristics and the additional input coming from the measures of the Additional Protocol.

3.3. Coverage of Acquisition Paths: how to avoid too much redundancy?

"Recognizing that weapons-usable nuclear material must be manufactured through a series of

discrete steps, the Secretariat has identified, described and characterized every known process for carrying out those steps. Any possible route from source material to weapons-usable material is called an acquisition path. Under integrated safeguards, all plausible acquisition paths by which a State might seek to acquire nuclear material for a nuclear explosive device will be covered by safeguards measures. The integrated safeguards approach for a State will therefore be designed to provide coverage of acquisition paths involving diversion of declared nuclear material from different stages of the nuclear fuel cycle, and to address clandestine routes to the acquisition of weapon-usable nuclear material. For any path where both undeclared activities and diversion from declared activities would be involved, the coverage will include verification measures on nuclear material in the declared activities as well as measures that would be needed to detect the undeclared activities."

The principle to cover all plausible acquisition paths seems logical. It is not clear how the IAEA is going to evaluate whether an acquisition path is plausible or not; what is clear however is that it is a matter of judgement. Can an entirely clandestine route constitute a plausible path? What is the importance of capacity, capability, skills, experience, past experience, licenses, etc. present in a State?

An acquisition path typically includes a number of elements or segments. The need to impose the coverage of both the declared segments and the undeclared segments of the path is therefore unclear, provided that enough segments are appropriately covered, so that credible assurance can be obtained that potential diverters that have chosen this acquisition path will be detected.

This requirement is interpreted in the LWR proposal as follows: *"The acquisition path that includes the production of undeclared Pu in a reactor is rightly said to be covered by the measures of the Additional Protocol. These measures provide an opportunity to detect the acquisition of the source material and the clandestine reprocessing of the Pu produced in the reactor."* The above boundary condition requires however that additional activities need to be carried out in the reactor itself on top of the traditional measures related to Nuclear Material Accountancy and regular PIV. This was translated in the approach by the introduction of unannounced inspections and core sealing associated with surveillance during core opening. These measures may be difficult to implement in certain circumstances (Unannounced Inspections) and/or have a direct impact on costs (surveillance during core opening), while they do not seem to be necessary as the path is addressed by Additional Protocol measures in a way that should give a credible assurance.

3.4. Non Discrimination Between States: how to implement it?

"The same overall objective and basic principles of integrated safeguards will be applied in a non-discriminatory manner in all States. Verification of declared nuclear material at facilities will be based on model integrated safeguards approaches developed for specific facility types to ensure consistency. Through the application of the supporting guidelines and criteria developed for integrated safeguards, similar procedures will be used in all States. However, the integrated safeguards approach for an individual State will take into account specific features and characteristics of the State and its nuclear fuel cycle to optimise effectiveness and efficiency, but again in a non-discriminatory manner."

Non-discrimination means that all States are treated equally. In classical safeguards it is translated into rules which were mainly based on the quantities and types (sensitivity) of nuclear materials. Or, the more material there was in a State, the higher the potential for diversion and the more verification activities needed to be performed by the IAEA in each type of facilities. The Additional Protocol introduces the use of "qualitative" measures. In integrated safeguards, due to the large set of measures available, equal treatment cannot be translated by such a "simple" quantitative / mechanical manner. It should be tailored to each country, based on a common set of rules identified by the IAEA, and not relying solely (or mostly) on the amount of nuclear material.

The facility approaches developed for Integrated Safeguards remain basically mechanistic and linked to the same quantitative parameters as in the classical safeguards system. It is not yet clear what specific features and characteristics of the State's nuclear fuel cycle are meant and how they can change (further reduce?) the IAEA activities in the facilities, whether classical safeguards driven or Additional Protocol driven. It is not clear how a State or an operator will benefit (and we agree all should benefit in a similar way) from being more open and demonstrating transparency

and a responsible behaviour towards the IAEA.

One should investigate seriously how other less quantitative parameters could be used when determining the effort in facilities. Elements could be: enhanced follow-up between nuclear fuel cycle facilities, transparency towards the IAEA, direct access to the relevant operational data set, other non quantitative even not fully-authenticated data that confirm the operational status of the facility, etc.

In a related move, an attempt should be made to include in the facility safeguards approach the use of "qualitative" measures. Taking the full extent of this means that, in integrated safeguards, the allocation of resources to verification of non-diversion of nuclear material in a given type of facility should be not primarily driven by the quantity of nuclear material. This would open the way to significant reductions in the classical safeguards measures. As a matter of illustration, when integrated safeguards are implemented, the intensity of verification for non-diversion of material of a 30 t fuel fabrication plant and a 100 t fuel fabrication plant may not be proportional to the size of the plant; the concept of timeliness goal remain valid, but it could be based on more elaborate rules than only the quantity of nuclear material. The concept of "safeguards in depth" could be generalised to this effect.

The application of some of the measures of the Additional Protocol such as access right to buildings on site can provide a further element of confidence in the overall assessment, leading to an "integrated facility approach»: combining at the facility level the verification measures of non diversion of nuclear material and the Additional Protocol measures.

When properly worked out, also these more qualitative parameters and integration work could form the basis for a process that meet the stated objective of non-discrimination.

3.5. Recoverability within existing Timeliness Goals: consistent with IS philosophy?

"Verification procedures for specific types of nuclear material are modified in a way that would allow the Agency to re-establish the inventories of those materials within the applicable traditional timeliness period if indications of possible diversion or undeclared nuclear material or activities come to light."

This requirement is difficult to justify and appears as an overly conservative application of the more global requirement " If during the implementation of Integrated Safeguards, the Agency were not able to reaffirm the conclusion …. Corrective actions would have to be taken. Depending on the circumstances, this could involve a range of action including restoring traditional safeguards activities…".

The current requirement means for instance that if the IAEA cannot conclude anymore on the absence of clandestine material or activities in State for whatever reason, the nuclear material inventory in all LWR reactors in the State needs to be re-established within 3 months, rather than within the Integrated Safeguards timelines goal of 12 months.

The consequences of this requirement are translated in the proposed safeguards approach by the need to have open core surveillance and reactor sealing during operation. These are resource intensive requirements requiring one to two additional visits to each LWR per year. They allow recovering the inventory within three months. If these measures would not be implemented, the inventory would be recovered at the next open core PIV. So at maximum, it reduces the time for reestablishment by 9 months for the reactor under typical annual refuelling, and does not prevent core sealing in a short time in case of doubt.

It could easily be argued that if the IAEA cannot conclude anymore about the absence of undeclared material and activities in a State, the IAEA should better use its limited resources to investigate and resolve the serious anomaly rather than performing a crash action to restore the inventories in all LWRs within three months.

And if the IAEA happens to have a clear indication of diversion or of undeclared activities in a State, then it should trigger an effective response mechanism at the international level, which would be much broader than increased classical safeguards activities.

3.6. Implementation issue: meaningful use of Complementary Access

The use of complementary access is not a boundary condition, but its implementation has direct resource implications. The number, the frequency and the activities performed during Complementary

Access play a role in the overall effectiveness and efficiency as well as to the perceived additional burden to the State and the operator. We do not question the need for the IAEA to check the completeness and correctness of the declaration. However, there is always a risk of becoming overly systematic, which can lead to a wrong allocation of efforts. Facilities are different and should be assessed against their potential role in covering undeclared activities; guidelines should be derived to direct the level of efforts depending on the sensitivity of the facilities. Here again, programming of Complementary Access should be based on qualitative factors more than on quantitative targets.

Those guidelines should be implemented consistently, in a non-discriminative way, and with due account taken of the State-specific situation. For instance, going to 1 mine per year in country A and in country B can be considered discriminatory if the number of mines is not the same, and may be a misdirected allocation of resources in terms of effectiveness as well as efficiency. The same way, calling systematically a Complementary Access at any visit at a research facility may also be a wrong allocation of resources and possibly send a wrong signal to facilities operators and States on the benefit of signing Additional Protocol. The amount of inspection resources spent in a given country under full scope safeguards and under integrated safeguards has no reason to be the same: the global expectation is that it should be less, but of course it can also be more if the situation warrants.

4. Suggestions to ensure a virtuous circle

The restrictions imposed by these boundary conditions on integrated safeguards will entail consumption of IAEA resources, and will cap the possible savings. As a matter of fact, the IAEA estimates the total savings in the range of 5% of PDI and in the range of 20% for safeguarding of reactors and LEU fuel fabrication plants in countries where Integrated Safeguards will be implemented. It is certainly lower than what was expected by many operators and member states. In terms of the allocation a fixed amount of IAEA resources, it means fewer resources available to the Additional Protocol related activities, even though they are expected to bring a higher yield. It unfortunately leads to an over-weighted allocation towards the classical measures and an under-weighted allocation towards Additional Protocol measures, resulting in an under-optimised overall system in terms of both effectiveness and efficiency.

It will also lead to the risk of not implementing properly the Additional Protocol measures and also not getting the necessary feed back to assess their actual effectiveness. We would therefore enter in a vicious circle: less resources on AP means less credibility on uncovering clandestine activities means less trust means more emphasis on “classical safeguards” ……

While we would expect a virtuous circle: enough resources for Additional Protocol related activities means more confidence on the absence of undeclared activities means less use of resource-consuming “classical safeguards”.

The IAEA could:

1 First independently assess the resources needed, on a steady state, to implement properly the AP measures
2 It will then give a target to the level of resources expected to be allocated on the verification of non-diversion of nuclear material
3 Estimate the total effectiveness of the combinations of the measures, which has to be more than the effectiveness of full scope safeguards only
4 If not, start the process again, with first a reallocation of resources within AP measures
5 If yes, start the process again in view of further increasing the overall effectiveness

This is the target for the steady state system. We recognise that, at this time, we do not have enough feed back from the implementation of Additional Protocol measures to get an estimate of their effectiveness and to correlate the level of resources devoted to these measures with their effectiveness. We expect though that enough resources can be devoted from the beginning to AP

measures to check their “value”. And if the experience demonstrates that their value is not as much as we expect, it will bring a rationale for re-opening the discussion on the value and effectiveness of the Additional Protocol.

We would like also to highlight that the IAEA safeguards budget is applied across the whole safeguarded countries. It is not discriminatory that, for a given country, resources allocated to inspection actually decrease when the total combination of measures brings the same or a higher level of effectiveness. And it will allow reallocating some of the resources saved from Integrated Safeguarded States towards less transparent States.

Conclusion

The existing boundary conditions for the conceptual framework of integrated safeguards have led to estimated savings in the classical safeguards area of around 5% of the present IAEA Person Days of Inspection. This is largely insufficient for the IAEA to perform all the measures of the Additional Protocol and to strengthen safeguards also in non- Additional Protocol Member States. We have argued in this paper that re-opening the discussion on some or all of the boundary conditions for the Integrated Safeguards framework may save additional resources without touching the overall performance of the IAEA safeguards system or the level of assurance the IAEA can obtain on the absence of a diversion and clandestine activities in a state.

The first and unquestionable added value of the Additional Protocol is the enhanced access to information and location given to the IAEA. Then it is a matter of implementation. We believe that, from the beginning, adequate resources must be available to estimate the value of the measures of the Additional Protocol. If experience showed that the value is not high enough, we must have the courage to conclude that the Additional Protocol in its present form is not the right tool and that it should be revisited.

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Reference Materials for Destructive Analysis in Nuclear Safeguards

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1. Introduction

Accountancy and control of nuclear material require analytical measurements that "shall either conform to the latest international standards or be equivalent in quality to such standards"[1]. This applies in particular for the quantitative verification of nuclear material as used in traditional safeguards, but it also applies to other measurements as used, for instance, in environmental sampling. Measurement standards are an indispensable tool wherever measurements are carried out. Their fundamental role is to establish traceability of a measured value (i.e. the analytical result) to a primary unit of measurement as defined in the SI system. Only measurement results that are traceable to a common reference namely the respective SI unit (by an unbroken chain of comparisons) can be regarded as truly comparable. In measurements of amount of material, these measurement standards are generally provided in the form of reference materials (RM). Such a reference material shall consist of "a material or substance which is homogeneous and for which one or more values are well established"[2]. Reference materials serve for calibration of a measurement instrument, for validation of a measurement technique, for the periodic assessment of a measurement system or for the assignment of values to materials.

Traditional safeguards involves "a quantitative verification of the accountancy of fissile material by independent verification measurements"[3]. This implies the comparison of two (or more) independently obtained measurement results. As mentioned above, true comparability is only possible if the reference materials used in the different measurement procedures provide traceability to a common reference. This is an essential precondition when evaluating results (typically paired data obtained from the same item) for safeguards purposes.

In the context of nuclear material safeguards, specific nuclear reference materials need to be available. These contain well known amounts of isotopes or elements relevant to the nuclear fuel cycle. In the following sections we will discuss the different levels of certification of reference materials, the particular challenges in their preparation and their application. This discussion will be complemented by a list of available reference materials suitable in the context of accountancy and verification, and the respective supplier information.

2. Reference Materials

The "International Vocabulary of Basic and General Terms in Metrology[4] provides definitions that are useful to recall in this context. Most of the definitions are taken from ISO Guide 30:1992[5]. In

order to illustrate the definitions we have added examples taken from the nuclear area.

*A **primary standard** is a standard that is designated or widely acknowledged as having the highest metrological qualities and whose value is accepted without reference to other standards of the same quantity.*

High purity metals as used in chemical analysis for element assay (e.g. uranium metal or plutonium metal) are typical examples for primary standards.

*A **secondary standard** is a standard whose value is assigned by comparison with a primary standard of the same quantity.*

Compounds of uranium or plutonium as used in chemical analysis for element assay (e.g. uranium dioxide pellets or plutonium oxide) are typical examples for secondary standards. Their reference value has been assigned by comparison with the respective metal standard.

*A **working standard** is a standard that is used routinely to calibrate or check material measures, measuring instruments or reference materials.*

Mixed solutions of uranium and plutonium as used for checking the validity of the calibration of hybrid K-edge instruments (applied for the assay of U and Pu in reprocessing input solutions) are typical examples for working standards. Materials that are metrologically equivalent to working standards or secondary standards are frequently used in inter-laboratory comparisons or external quality control programs.

*A **certified reference material** (CRM) is a reference material, accompanied by a certificate, one or more of whose properties are certified by a procedure which establishes traceability to an accurate realization of the unit in which the property values are expressed and for which each certified value is accompanied by an uncertainty at a stated level of confidence.*

As outlined above, reference materials serve for calibration of measurement instruments, for the assessment of a measurement method, for assigning values to materials or for long-term quality assurance. They may be in the form of pure or mixed gases, liquids or solids. For safeguards-related applications, reference materials are available with known element content (elemental reference material), known isotope content (spike reference material) or with known isotope ratios (isotopic reference material), normally involving thorium, uranium or plutonium and less frequently americium, neptunium or curium.

2.1 General Challenges in the Preparation and Certification

Reference materials (RMs) serve as the basis for millions of measurements performed each year. The “reliability” of these materials, the associated reference value and the stated uncertainty are of key importance to the laboratories and ultimately the to users of the measurement results. Particularly in nuclear material accountancy and control, where important decisions are based on such measurement results, a high degree of confidence needs to be put in the measurements and in the reference materials. Consequently, the preparation and certification of reference materials is a very demanding task. It requires a thorough understanding of the processes involved and a very careful execution of the preparative and analytical work in the laboratory. Only specialized laboratories and skilled operators will be able to perform this task properly.

The preparation of nuclear reference materials is also associated with handling problems. Their preparation needs to be carried out in glove-boxes which is often not compatible with the requirement for highest possible accuracy or best possible working conditions.

All reference materials are supplied with certificates which apply to the particular reference material. The certification is carried out on a set of sub-samples of the reference material, yet it has to be applied to all individual units produced from the entire batch. Great care has therefore to be taken that the certified values apply to the individual units of the reference material, hence during the certification process proof of homogeneity will need to be provided.

Isotopic reference materials have their particular problems. Isotopic contamination during handling is an ever-present possibility. For this reason the reference material is manipulated in clean environments – new or dedicated glove-boxes wherever possible – and a careful check is kept on the isotopic ratios before and after manipulation. This implies a large number of validation measurements. Such considerations are magnified when IRMs suitable for environmental analysis are being certified and dispen-

sed. Small contamination from the laboratory environment can invalidate the isotopic ratios of the reference material and this must be eliminated – and shown to be eliminated.

Primary reference materials are rare and expensive and are commonly used to certify secondary reference materials. This certification process is a comparison in principle and given the nature of measurements, a verification is always carried out against known reference materials and preferably using a separate technique to show up possible sources of bias.

The values and their uncertainties for certified reference materials are calculated following the ISO Guide for the Expression of Uncertainty in Measurement (GUM)[6]. This demands a good knowledge of the sources and magnitudes of uncertainties during the measurement process. If a second measurement method is also applied for certification, then not only the uncertainties arising during this method have to be quantified but also the degree of overlapping between the methods need to be expressed as correlations. Often it is easier conceptually to certify using one measurement method and verify using the second. This is done especially for the cases where the uncertainties of the two methods are considerably different in magnitude. In this case the method with the smaller uncertainties will be applied for the certified values and the second method considered as an independent verification.

2.2 Elemental Reference Materials

An Elemental Reference Material is one for which the amount (preferably to be expressed in moles) or the amount content (expressed in moles g^{-1} or in the derived quantity $g \cdot g^{-1}$) are certified. Typically these can be pure metals or other solid samples such as oxides or also as solutions with certified element amount content. Although in principle the isotopic abundances of the material are not certified it is useful if they are specified. Although such materials usually consist of long-lived isotopes, corrections have to be steadily applied for losses through radioactive decay.

Elemental Reference materials are typically used to calibrate methods such as titration, coulometry or K-edge absorptiometry. They are also valuable for providing the traceability link when certifying isotopic spikes. In this case, the isotopic abundances of the material have also to be measured and certified.

2.3 Isotopic Reference Materials

An Isotopic Reference Material (IRM) is a material normally of one element, but can also in principle be multi-elemental, certified for isotope abundance ratios. In practice, the isotopic ratios of all isotopes to the most abundant of the element are certified. Following this, the isotope abundances can readily be calculated and certified. It should be noted that it is much preferable to define and use the isotopic ratios as primary certified values as these are not correlated - in contrast to isotopic abundances which can be severely correlated and lead to incorrect assessments of uncertainties when applied directly.

IRMs are applied to calibrate mass-spectrometers. The mass-fractionation of the mass-spectrometer is calculated by measuring one or more IRMs and this value is then applied to the calculation of corrected isotopic ratios on samples measured under similar measurement regimes. Alternatively, the IRM is used as a quality control sample by being measured in the same measurement process as the samples and using the measured value of the isotopic ratio to control and assure the measurement of the isotopic ratios in the samples.

Calibrating a mass-spectrometer in this way implies that there are three main contributions to the final uncertainty of the value of an isotopic ratio: the uncertainty from the certification of the IRM, the uncertainty from the measurement of the IRM and the uncertainty from the measurement of the sample. It therefore requires that the certified isotopic ratios of the IRM have very small uncertainties.

2.4 Spike Reference Materials

A Spike Reference Material is an IRM certified for isotopic amount and amount content. Typically these are isotopically enriched materials such as uranium enriched in ^{233}U or plutonium enriched in ^{244}Pu or in ^{242}Pu. The amount or amount content of the enriched isotope is certified (expressed in

moles·g^{-1} or in the derived quantity g·g^{-1}). Also required are the isotopic ratios of all other isotopes relative to the highest abundance isotope (normally called the 'spike isotope'). Spike reference materials are applied almost exclusively for the measurement of amounts of nuclear materials by isotope dilution mass-spectrometry (IDMS). Most Spike reference materials are therefore relatively highly enriched (>90%) in the spike isotope as this minimizes the uncertainty in the calculated value of the amount of nuclear material measured.

There are several parameters that influence the final uncertainty of the amount of nuclear material measured by IDMS and the enrichment of the spike is only one of these. In fact successful IDMS measurements can be made with partially enriched RMs: so-called Large Size Dried Spikes of uranium and plutonium with a 235U spike enriched to 20% are applied with very good success. In fact, all parameters of the spike have to be certified with low uncertainties for successful application in IDMS.

For measurements of environmental samples, it is rare that the absolute amount of nuclear material has to be measured accurately. The isotopic vector of the nuclear material is often the most critical piece of information to be measured. For such measurements, a very highly enriched spike is a great advantage. If the enrichment is sufficiently high, the isotopic ratios as well as the amount of material of the sample can be measured in a spiked sample. The use of a spike brings with it the added advantage that a known amount of the uranium or plutonium is introduced into the purification chemistry, reducing losses and also providing a standard signal size in the mass-spectrometer.

3. Available Reference Materials

A large number of reference materials are available on a commercial basis. The ESARDA Working Group on Standards and Techniques for Destructive Analysis compiled relevant information on nuclear reference materials as used for destructive analysis. Significant input was provided by the IAEA, who organizes a respective Consultants Group Meeting at regular intervals. The information is presented in three tables, summarizing information on elemental reference materials, isotope ratio reference materials and spike reference materials and various others, respectively. The list of reference materials presented in the tables below is intended to provide an overview on the spectrum of available reference materials from the major suppliers. The list is not intended to be fully exhaustive. Up-to-date information on the availability of reference materials and their exact specifications is available from the suppliers. Values given in the tables below are therefore only of indicative nature.

The number of suppliers of nuclear reference materials is limited due to the specific handling requirements imposed on nuclear materials and obviously due to the limited market for such materials. Suppliers that were considered for the present compilation are:

- Institute for Reference Materials and Measurements (EC, IRMM), 2440 Geel, Belgium; http://irmm.jrc.cec.eu.int/
- New Brunswick Laboratory, (NBL), 9800 S. Cass Ave., Argonne IL 60439, USA; http://www.nbl.doe.gov/
- CETAMA, CEA/Valrho-Marcoule, DCC/DRRV, BP 171, 30207 Bagnols sur Ceze, France
- V.G. Khlopin Radium Institute, 2-nd Murinsky Av. 28, 194021 St. Petersburg, Russia
- Ural Electrochemical Integrated Plant (UEIP), Materials are available through TENEX, http://www.tenex.ru/izotop_en.html#uran

Materials that are no longer commercially available, either because the initially produced stocks ran out or because the supplier(s) went out of business, are indicated in brackets.

4. Conclusions

Nuclear reference materials are the essential pillar in establishing traceability and enabling comparability of measurements of nuclear material. Without a solid common metrological basis, the comparative evaluation of measurement data produced in different laboratories will be impossible. Such a

Table 1: Commercially Available Reference Materials for Element Assay

Use	Material Description	Supplier Code	Certified Element	Accuracy [%]	Unit Size
Redox Titr.	Potassium Dichromate	NIST 136e		0.005	60 g
U Assay	Nat-U Metal	NBL 112a	U	0.017	26 g
		EC-101	U	0.005	1 g
		CETAMA MU2	U	0.005	0.5 - 1 g
	Depl-U Metal	NBL 115	U	0.005	75 g
	Depl-UO_2 Pellets	EC-110	U	0.015	25x1 g
		(UK-U1/80350)	U	0.007	50x1 g
	Nat-UO_2 Pellets	IRMM 106	U	0.03	15x10 g
		CETAMA OU1	U	0.102	10x0.4 g
	LE-UO_2-Pellets	NBL-125 a	U	0.014	5.4 g
	Natural U_3O_8	prev. NBS 950	U	0.02	25 g
		CETAMA OTU1	U	0.1	25 g
		(NBL 129)			
	Nat-UO_3	(NBL 18)			500 g
	Nat-UF_4	(NBL 17b)			200 g
	Ore	CETAMA MIN A - MIN F	U		6x100 g
	Nat-UNH solution (10mg/mL)	NBL 145	U	0.011	20 mL
	HE-UNH solution	NBL 135	U	0.18	
	LE-UF6	NBL 113b	U	0.033	1700 g
	Pitchblende Ore/Dunite Mix	NBL 42a	U	0.17	100 g
				0.16	100 g
				0.17	100 g
				0.44	100 g
	Phosphate Rock Ore	NBL 1a	U	0.38	50 g
	Low-Grade Pitchblende	NBL 3b	U	None stated	100 g
	Carnotite Ore	NBL 4	U	None stated	100 g
	Carnotite Ore	NBL 5	U	None stated	100 g
	Pitchblende Ore/Silica Mix	NBL 101a	U	1.3	50 g
	Pitchblende Ore/Silica Mix	NBL 102a	U	1.8	50 g
	Pitchblende Ore/Silica Mix	NBL 103a	U	1.4	50 g
	Pitchblende Ore/Silica Mix	NBL 104a	U	2.0	50 g
	Pitchblende Ore/Silica Mix	NBL 105a	U	2.0	50 g
U/Th Assay	HEU+Th Carbide BISO	(NBL 118)	U ^{235}U Th	0.02 0.0014 0.012	10 g
	HEU+Th Carbide TRISO	(NBL 119)	U ^{235}U	0.01 0.0014	15 g
	Monazite Sand/Silica Mix	NBL 106a	Th	0.29	
		NBL 107a	Th	0.19	
		NBL 108a	Th	0.39	
		NBL 109a	Th	0.86	
		NBL 110a	Th	0.96	
Pu-Assay	^{239}Pu metal	(NBL 126)	Pu	0.018	1 g
		CETAMA MP2	Pu	0.04	0.5 - 1 g
	^{239}PuGa Alloy	(UK-Pu1/80990)	Pu	0.017	0.7 g
	^{239}PuO2	IRMM 210	Pu	0.05	1 g
		NBL 122	Pu	0.039	1 g
U/Pu Assay	*LWR (U,Pu)O_2 pellets*	*CETAMA-CETAMOX*	*U* Pu	*0.03* 0.05	
Np Assay	Np oxide microspheres	IRMM-502	Np		100mg
	^{237}Np solution	(AEA)	Np	0.9	5 g
		LMRI		1	
Am Assay	^{241}Am Solution	NBL-132	Am		5 g sol.
		LMRI	Am	0.8	1 g sol.

Table 2: Commercially Available Reference Materials for Isotope Ratio Measurements

Use	Material Description	Supplier Code	Certified Ratio	Accuracy [%]	Unit Size
U Isotopic	$^{233/235/238}$U solutions (various isotope ratios)	(IRMM-072) IRMM-073	235/238 233/238	0.03 0.03	15 x 1 g
	$^{233/235/238}$U solutions (1:1:1)	NBL-117	233/235/238	0.06, 0.09	0.3 g
		EC-199	233/235/238	0.03, 0.02	5 g
		(UK-U2/96059)	233/235/238	0.04, 0.05	1 g
		(UK-U2/96058)	233/235/238	0.04, 0.05	1 g
	$^{233/236}$U nitrate	CETAMA MIRF 02	233/236	0.1	0.2 mg
	0.02-97% ^{235}U oxides	NBL 0002 – U970	235/238	0.05	1 g
		BNFL	235/238	0.05	1 g
		LMRI	235/238	0.05	1 g
	0.000 011 – 99.994 7% ^{235}U	TENEX (UEIP)	235/238	0.05	
	0.02-97% 235UF6	CBNM-21 / -27	235/238	0.05	7 x 20 g
		BNFL	235/238	0.05	5-10 g
		LMRI	235/238	0.05	
	LE-UO2 pellet (4% ^{235}U)	NBL 125a	235/238	0.05	5.4 g
	LE-UF6 (4.5% ^{235}U)	NBL 113b	235/238	0.05	1700 g
	93% ^{235}U oxide	NBL U930d	235/238	0.1	5mL (5mgU)
Pu Isotopic	$^{239/240}$Pu solutions (var. ratios)	(UK-Pu4/92136)	240/239	0.11	1 g
		(UK-Pu6/92028)	240/239	0.08	1 g
	$^{239/240/242}$Pu solutions (1:1:1)	(UK-Pu3/92134)	239/240/242	0.06, 0.1	1 g sol
		(UK-Pu3/92133)	239/240/242	0.06, 0.1	1 g sol
	$^{239/240/242/244}$Pu solution (3:3:3:1)	(UK-Pu5/92138)	239/240/242 244/239	0.11, 0.19 0.24	1 g sol
	$^{239/242}$Pu nitrate (various $^{239/242}$Pu ratios)	IRMM-290	239/242	0.01	7 x 1 mg
	$^{239/242}$Pu nitrate (1:1)	NBL-128	239/242	0.026	1 mg
		CETAMA MIRF 01	239/242	0.05	0.1 mg
	Pu-sulfates (various isotope ratios)	NBL-136 (prev. NBS-946)	240/239	0.12	0.25 g
		NBL-137 (prev. NBS-947)	240/239	0.12	0.25 g
		NBL-138 (prev. NBS-948)	240/239	0.12	0.25 g
	Pu oxide	NBL 122	240/239	0.12	1 g
U/Pu/Nd Assay, Isotopic	U/Pu/Nd-solutions	(UK-UPuNd1/90085) (LWR)	U, Pu, Nd		1 ml
		(UK-UPuNd2/90088) (FBR)	U, Pu, Nd		1 ml
Am Isotopic	$^{241/243}$Am nitrate solution	NBL-134			
Nd Isotopic	Nd seven isotope mix. (2 µg Nd/g)	(AEA RM W10930)	Nd	0.04 - 0.11	1 g

comparison is, however, a key element in the process of safeguarding nuclear material. The continuous provision of adequate and reliably certified nuclear reference materials is an important task for few highly specialized metrolgical laboratories. Measurement laboratories are strongly encouraged to benefit from the correct use of certified reference materials for calibration, method validation and for quality control purposes. As chemical analytical methods often provide the basis for radiometric or non-destructive techniques, a link between the latter and reference materials for destructive analysis can be established, thus contributing to establish traceability of non-destructive measurements.

Table 3: Commercially Available Spike Reference Materials

Use	Material Description	Supplier Code	Certified Quantity	Accuracy [%]	Unit Size
U Tracer	^{233}U solution (0.5 mg ^{233}U/g)	NBL-111a	^{233}U	0.025	10 g
		(NBL-111) (prev. NBS 995)	0.05	10 g	
	^{233}U solution (1 mg ^{233}U/g)	IRMM-040a	^{233}U	0.18	10 g
	^{235}U metal (93%) ^{235}U solution (7 mg ^{235}U/g)	(NBL-116) NBL-135	U, ^{235}U	0.007, 0.005	1.5 g
		(prev. NBS-993)	U, ^{235}U	0.18, 0.0013	100 mg
	^{235}U solution (1 mg 235U/g)	NBL U930d	^{235}U	0.0049	5.4 g
	^{235}U solution (1 µg 235U /g)	IRMM-050	^{235}U	0.025	10 g
	^{235}U solution (2.4 µg 235U /g)	IRMM-051	^{233}U	0.18	5 g
	^{238}U solution (1.3 mg 238U/g)	IRMM-052	^{238}U	0.05	5 g
	^{238}U solution (50 µg 238U /g)	IRMM-053	^{238}U	0.05	5 g
	^{235}U solution (1 mg 235U /g)	IRMM-054	^{235}U	0.03	5 g
	^{238}U solution (1 mg 238U /g)	IRMM-056	^{238}U	0.03	10 g
	^{233}Usolution (1 mg 233U /g)	IRMM-057	^{233}U	0.13	10 g
	^{233}U solution (1 ng 233U /g)	IRMM-058	^{233}U	0.13	10 g
Th Tracer	^{230}Th solution (40 µg 230Th/g)	IRMM-060	^{230}Th	0.74	10 g
Pu Tracer	^{242}Pu solution (9 µg 242-Pu/g)	IRMM-044	^{242}Pu	0.15	10 g
	^{242}Pu solution(89 µg 242Pu/g)	IRMM-049c		0.14	10 g
	^{239}Pu solution (115 µg 239Pu /g)	IRMM-082	^{239}Pu	0.06	5 g
	^{239}Pu solution (12 µg 239Pu /g)	IRMM-081	^{239}Pu	0.06	5 g
	^{240}Pu solution (0.82 mg 240Pu /g)	IRMM-083	^{240}Pu	0.06	1 g
	^{244}Pu solution (9 ng 244Pu /g)	IRMM-084	^{244}Pu	0.21	5 g
	^{242}Pu solution (9 ng 242Pu /g)	IRMM-085	^{242}Pu	0.15	5 g
	^{242}Pu dried nitrate	NBL-130	^{242}Pu		1 mg
	^{244}Pu solution (0.9 µg 244Pu /g)	IRMM-042a	^{244}Pu	0.2	10 g
	^{244}Pu dried nitrate	(NBL-131) (prev. NBS-996)	244Pu	0.15	1 mg
	$^{240/242/244}$Pu dried nitrate	(NBL-144)			
U/Pu Tracer	$^{235U/239}$Pu dried nitrate (LSD)	(IRMM-1027 a, b, c, d, e)	^{235}U, U ^{239}Pu, Pu	0.1, 0.1 0.1, 0.1	50 mg U + 1.5 mg Pu
		IRMM-1027f, g	^{235}U, U ^{239}Pu, Pu	0.1, 0.1 0.1, 0.1	50 mg U + 1.5 mg Pu
	$^{233U/242}$Pu solution (1 mg ^{233}U + 0.1 mg ^{242}Pu / g)	IRMM-046b	^{233}U, ^{242}Pu		10 g
Am Tracer	^{243}Am	NBL-133 (Harwell 95/243/37)	^{243}Am	0.7	1 g

Dissemination of information on available reference materials and an intensive dialogue between the users of such materials and their producers will be a key element for an efficient use of resources and of precious base materials. The continued exchange of information from field-experience in the use of reference materials and from the thorough metrological understanding of the measurement process and the associated certification procedure in a forum like the ESARDA WGDA will be highly beneficial for either side.

On the other hand, the producers of reference materials are encouraged to continue their co-operation and assure the world-wide availability of certified reference materials for the analysis of nuclear materials.

Table 4: Commercially Available Reference Materials for Impurities

Use	Material Description	Supplier Code	Unit Size
Impurities	U_3O_8	NBL-123	7x25 g/set
		NBL-124	7x25 g/set
		(NBL-95)	6x25/set
		(NBL-98)	6x25/set
		(NBL-114)	100 g
		CETAMA Agaric	25 g
		CETAMA Chanterelle	25 g
		CETAMA Bolet	25 g
		CETAMA Morille	25 g
	U-metal	CETAMA Floralies	17 x 20 g
		(NBL 112)	50 g
	U-concentrates	CETAMA (A-H)	8 x 50g
	Nat-UO_3	(NBL 18)	500 g
	Nat-UF_4	(NBL 17b)	200 g
	ThO_2	NBL-66	7x25 g/set

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An overview of the comprehensive Test Ban Treaty

Nuclear-Test-Ban Treaty Organization

The adoption of the Comprehensive Nuclear-Test-Ban Treaty (CTBT) by a special session of the 50th United Nations General Assembly on 10 September 1996 reflected almost four decades of international efforts designed at bringing an end to nuclear testing.

By banning all nuclear test explosions, the CTBT constrains the development and qualitative improvement of new types of nuclear weapons. The Treaty thus contributes effectively to the prevention of both horizontal and vertical proliferation, and to the process of nuclear disarmament, which continue to be priority objectives of the international community in the field of disarmament and non-proliferation.

The Treaty's implementation is essential to the global nuclear non-proliferation regime. Article I of the Treaty prohibits any nuclear test explosion or any other nuclear explosion. States Parties also commit themselves to refrain from causing, encouraging, or in any way participating in the carrying out of any nuclear weapon test explosion or any other nuclear explosion.

Since monitoring is crucial to a nuclear test ban, the CTBT provides for an extensive verification regime, which consists of:

A. The International Monitoring System (supported by the International Data Centre and the Global Communications Infrastructure)
B. A consultation and clarification process
C. On-site inspections
D. Confidence-building measures

In order to implement these verification measures, the Preparatory Commission for the Comprehensive Nuclear-Test- Ban Treaty Organization (CTBTO Prep Com) has been established in Vienna. By providing for the cooperative establishment and operation of a global remote sensing network for international treaty monitoring, the CTBT is a unique treaty in the field of multilateral arms control.

The main purpose of the International Monitoring System (IMS) is to monitor the Earth in order to detect and provide data on possible nuclear explosions and ambiguous events. It uses seismic, infrasound, hydroacoustic and radionuclide monitoring technologies to register vibrations underground, in the air and in the sea, and to detect radionuclides released into the atmosphere by nuclear explosions. Once fully established, the IMS will be capable of detecting nuclear explosions of very low yield detonated in any environment on Earth.

The IMS consists of a global network of 337 monitoring facilities (170 seismic stations, 11 hydroacoustic stations, 60 infrasound stations, 80 radionuclide stations and 16 radionuclide laboratories). Many stations are located in remote areas in order to provide global coverage. This has presented logistical and engineering challenges unprecedented in the history of arms control.

The seismological component of the monitoring system detects and locates seismic events. Seismic data allow for the distinction between an underground nuclear explosion and an earthquake.

Hydrocaoustic monitoring detects acoustic waves produced by natural and man-made phenomena in the oceans. The data from these stations are used to distinguish between underwater explosions and other phenomena such as submarine volcanoes and earthquakes, which also propagate acoustic energy into the oceans. Only 11 stations are required because of the very efficient propagation of acoustic energy in the oceans.

The infrasound network uses microbarometers to detect very low-frequency sound waves in the atmosphere produced by natural and man-made events. Infrasound data are used to locate and distin-

guish between nuclear explosions, natural phenomena such as exploding volcanoes and meteorological events, and man-made phenomena such as space debris, and aircraft in supersonic flight.

The radionuclide network uses air samplers to detect radioactive particles released from atmospheric explosions or vented from underground or underwater explosions. IMS radionuclide laboratories analyze air samples for radionuclide materials that may have been produced by a nuclear explosion. The presence of specific radionuclides provides unambiguous evidence of a nuclear explosion. The presence of noble gases is particularly important in detecting releases from underground explosions. Half of the stations in the radionuclide network will also have the capacity to detect noble gases.

Since the Treaty opened for signature in 1996, significant advancement has been made in the establishment of the IMS. During 2002, progress was made on all aspects of installing monitoring stations in all four technologies. Site surveys for 88% of the total number of stations have now been completed. Altogether, 150 stations have been built or substantially meet specifications, of which 55 have been certified, a process that is necessary for the stations to become part of the IMS network. An additional 80 stations are currently under construction or in contract negotiations.

Some 80 facilities are already contributing data to the International Data Centre (IDC) in Vienna. The IDC is responsible for receiving, collecting, processing, analyzing and reporting on and archiving data from IMS stations and laboratories. The results of the work at the IDC are put together as different types of reports, usually referred to as IDC products. Data are processed immediately upon arrival at the IDC and the first automated products are released within minutes to States Signatories for further analysis and final judgement.

The Global Communications Infrastructure (GCI) provides communications links between IMS facilities and the IDC. The GCI is the first global satellite communications network to be based on Very Small Aperture Terminal (VSAT) technology. Monitoring facilities and States Signatories in all areas of the world can exchange data via their local VSAT earth stations.

The IMS network of sensors on land, under the oceans and in the atmosphere is already far more extensive, and sensitive, than anything one country can mount on its own. Such coverage is essential to deter potential violators and to detect them if necessary. If a State Party to the CTBT suspects that a nuclear explosion may have been carried out in violation of the Treaty, that State may request an on-site inspection (OSI) to clarify whether a nuclear weapon test or any other nuclear explosion has been carried out in violation of the Treaty. The OSI will also aim to gather facts, to the extent possible, which might assist in identifying any possible violator, thus serving as a final verification measure of the CTBT.

Before requesting an on-site inspection, States Parties are encouraged by the Treaty to try to resolve, either among themselves or with the assistance of the Organization, any matters which may indicate possible non-compliance with the CTBT's basic obligations.

The verification regime also includes confidence-building measures, which serve a twofold purpose. Firstly, they contribute to the prompt resolution of compliance concerns relating to chemical explosions. Secondly, they assist in the calibration of IMS stations by improving knowledge of how vibrations propagate through the Earth's structure, thus enhancing the accuracy of the location of seismic events.

The number of signatures and ratifications of the CTBT continues to increase, with Mauritania becoming the 100th State to ratify the Treaty on 5 May 2003. This figure signified an important milestone on the road to the universality of the CTBT. As of 25 June 2003, a total of 102 States had ratified the Treaty with more States expected to ratify prior to and during the forthcoming *Conference on Facilitating the Entry into Force of the CTBT* in Vienna from 3 to 5 September 2003. The Treaty has been signed by 167 States, indicating the support of the vast majority of governments for a verified end to nuclear testing.

A global environment where proliferation is prevented facilitates the disarmament process and encourages movement towards the ultimate elimination of nuclear weapons. By banning nuclear weapon test explosions and all other nuclear explosions in any environment, the CTBT represents an effective nuclear non-proliferation instrument and plays a crucial role in promoting global peace and security.

In Memoriam

Charles Beets

Charles Beets passed away in Liege in Belgium in December 2002 at the age of 80. He is survived by his wife Marie-Thérèse and his children Francois, Marie Christine and Claude.

In 1954 he joined the Belgium research centre at CEN/SCK in MOL.. Mol holds now, as it did then, a large number of different nuclear installations which offered various opportunities for the development and application of the new gamma and neutron detection techniques of that time. Charles was a man who could formulate elegant solutions to practical problems and successfully manage disparate groups of academics and researchers to conduct trials in real plant conditions. Such projects included safeguards on the Eurochemic reprocessing plant and on fuel fabrication plants. The R&D was performed in most cases in the framework of International Arrangements and co-operation with the ESARDA partners amongst these the the Euratom Safeguards Directorate and the International Atomic Energy Agency.

At that time, Charles recognised the importance of ESARDA as a vehicle and forum for European R&D, co-operation and harmonisation. In 1970, one year after ESARDA was formed, he negotiated the accession of CEN/SCK, and with it the Belgium nuclear industry, to the ESARDA association, where it has remained as an active member to this day. Within ESARDA he communicated and promoted his ideas for fabrication and reprocessing plant safeguards based on his famous MOL experiments.

He had a close relationship with the Belgium Government, and for the last two decades acted as scientific advisor to the Belgium Ministry of External Affairs. More specifically he was advisor to the Permanent Belgium Delegation to the IAEA on matters related to nuclear safeguards and non-proliferation and was a regular contributor at the IAEA in Vienna. Moving in these circles meant that Charles was always well informed, and not only about scientific and political matters.

For a long period Charles was the ESARDA program co-ordinator for Belgium and represented his country's interests as a member of the ESARDA Steering Committee even well into his retirement. Charles was a mild man, seldom angry but with a strong sense of fairness. He worked hard to ensure that the European non-nuclear weapon states were fully represented in advisory and opinion forming groups such as the many IAEA advisory and consultant groups at the begin of the NPT safeguards implementation.

Charles was heavily involved in Nuclear Safeguards, he was a humble and honest scientist who contributed greatly to the ESARDA organisation and will be fondly remembered by his colleagues.

A Word from the Secretary

The ESARDA website was launched in April 2002. It has proved to be very useful for providing information about the association and its events. It also features the most recent ESARDA bulletins.

It was particularly useful during the preparation for the Stockholm symposium held in May 2003. Participants were able to access abstracts (more than 140), the programme, registration and accommodation information. The conclusions of the symposium were added after the symposium.

The website has been visited more than 5000 times since it opened. This clearly demonstrates that the ESARDA community are enthusiastic about the website as an efficient way of finding information. Communication has improved and the Secretariat will continue to use this tool as a primary means of communication and for the dissemination of safeguards information.

The public is becoming more and more aware of nuclear issues and needs clear, easy to understand definitions of safeguards terminology. This information must come from authenticated, accurate, respected sources.

The SCCB is currently working on a document, which could appeal to the general public

The table shows everyday examples of reference material and the ESARDA equivalent.

ISSUE	EVERYDAY EXAMPLE	ESARDA RESPONSE
Definition of a word; verification of its sense	The pocket Oxford dictionary	ESARDA Glossary
Explanation of a technique; scientific principles	Encyclopaedia Britannica	Technical Sheets
Development, detailed description of concept, technique and implementation; history	Specialised literature	Course modules

Numerous scientific experts are currently working on a glossary of safeguards terminology that has clear and simple definitions. For a newcomer to safeguards, a clear and simplified explanation will be preferred to an exact in-depth definition and explanation which would be reserved for the specialist; so that newcomer can understand the terms without having to study the meaning in minute detail.

The IAEA glossary provides those in-depth definitions of safeguards terminology referred to above. Therefore it would be inefficient to re-create a similar document. However, a simplified reference document backed up by information taken from the IAEA glossary could be a useful resource. It would have the equivalent role of the pocket Oxford dictionary cited above. I take this opportunity to thank again the IAEA for agreeing that their glossary be used to construct and back up the ESARDA glossary.

The technical sheets would be more detailed explaining the basic concepts of various techniques used in safeguards.

Finally ESARDA intends to create a series of course modules drafted by safeguards specialists and university lecturers so that their knowledge can be passed on to students and new comers to the field.

The objective would be to put the knowledge at the disposal of universities and high level education institutions that have nuclear engineering courses.

Status of implementation:

The JRC has already drafted a preliminary version of the glossary that is under revision by the SCCB. It is expected that, within a few months a first version of the glossary will be available on the website. Improvement and expansion will then take place step by step. The ESARDA Working Groups are involved in this work.

As far as **Technical Sheets** are concerned, some “prototypes” have been drafted by SCCB members. They are being reviewed and ESARDA expects to put them on the web site at the same time as the glossary. The Working Groups will have significant input in the drafting of these Technical Sheets.

The principle of **course modules** has already been approved by the ESARDA Executive Committee. A small group is working on a possible structure for the modules. They are establishing contact with universities and ENEN (European Nuclear Engineering Network) in order to make sure that these courses are compatible with the European course programmes being developed under ENEN guidance.

The ESARDA Secretary will fully support these activities and will give his assistance as often as it is requested.