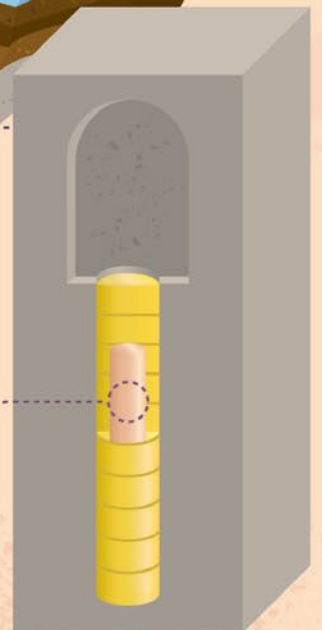
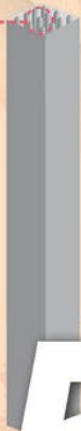
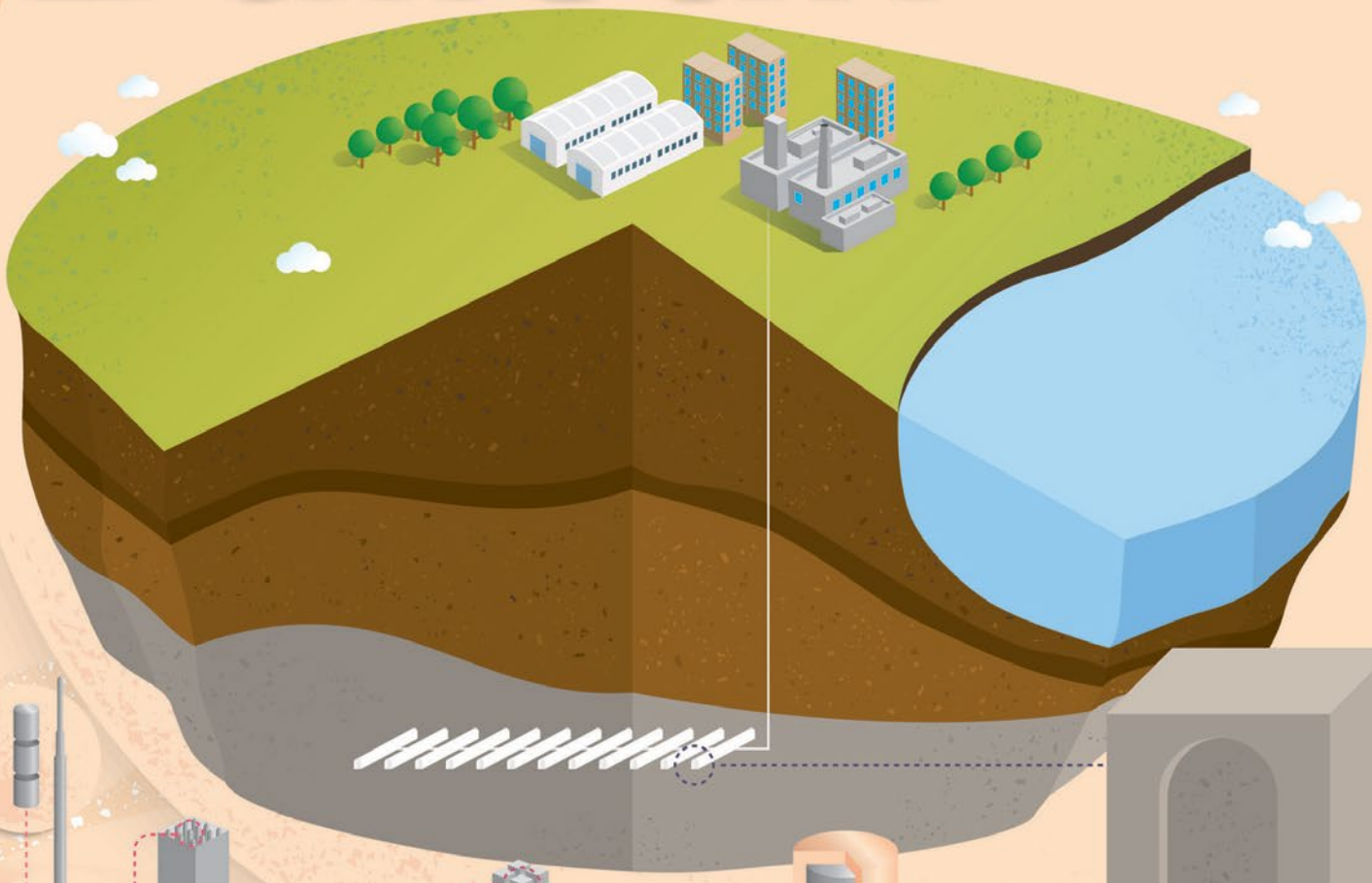


# Bulletin



N° **56**

**June 2018**

# Bulletin

## Table of Content Issue n° 56

ISSN 1977-5296

Number 56  
June 2018

### Editor

Elena Stringa

European Commission, Joint Research Centre,  
ITU - Nuclear Security Unit  
T.P. 800, I-21027 Ispra (VA), Italy  
Tel. +39 0332-786182  
EC-ESARDA-BULLETIN@ec.europa.eu  
Elena.Stringa@ec.europa.eu  
Hamid.Tagziria@ec.europa.eu

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

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

### Editorial Committee

K. Axell (SSM, Sweden)  
I. Popovici (CNCAN, Romania)  
A. Rezniczek (Uba GmbH, Germany)  
J. Rutkowski (FZJ, Germany)  
P. Schwalbach (EC, DG ENER, Luxembourg)  
F. Sevini (EC, JRC, G.II.7, Italy)  
Z. Stefánka (HAEA, Hungary)  
E. Stringa (EC, JRC, G.II.7, Italy)  
H. Tagziria (EC, JRC, G.II.7, Italy)  
J. Tushingham (NNL, United Kingdom)

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

Manuscripts are to be sent to the Editor (EC-ESARDA-BULLETIN@ec.europa.eu) following the 'instructions for authors' available on <https://esar.da.jrc.ec.europa.eu/> where the bulletins can also be viewed and downloaded.

Photos or diagrams should be of high quality.

Accepted manuscripts are published free of charge.

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

ESARDA Bulletin is published jointly by ESARDA and the Joint Research Centre of the European Commission and distributed free of charge to over 1000 registered members, libraries and institutions Worldwide.

The publication is authorized by ESARDA.

© Copyright is reserved, but part of this publication may be reproduced, stored in a retrieval system, or transmitted in any form or by any means, mechanical, photocopy, recording, or otherwise, provided that the source is properly acknowledged.

Cover designed by Laura Spirito, (JRC Ispra in Italy)

Printed by  
IMPRIMERIE CENTRALE, Luxembourg

### Editorial

O. Okko .....	1
---------------	---

### Peer Reviewed Articles

<b>GOSSER - Geological Safeguards and Security R&amp;D Project in Finland - How STUK prepares itself for the Final Disposal in Finland .....</b>	<b>2</b>
O. Okko, M. Moring, T. Honkamaa, E. Martikka, M. Hämäläinen	
<b>Non-destructive assay sampling of nuclear fuel before encapsulation.....</b>	<b>6</b>
C. Hellesen and S. Grape	
<b>Utility of Including Passive Neutron Albedo Reactivity in an Integrated NDA System for Encapsulation Safeguards .....</b>	<b>12</b>
S. J. Tobin, P. Peura, C. Bélanger-Champagne, M. Moring, P. Dendooven, T. Honkamaa	
<b>Ultrasonic Investigation of the Welding Area of Copper Canisters for Spent Nuclear Fuel ..</b>	<b>19</b>
C. Clementi, F. Littmann, L. Capineri	
<b>Usability of Monitoring at the Olkiluoto Repository Site for Safeguards.....</b>	<b>28</b>
E. Pentti and O. Okko	
<b>Elements of a Swedish Safeguards Policy for a Spent Fuel Disposal System .....</b>	<b>36</b>
G. af Ekenstam, L. Hildingsson, R. Fagerholm, C. Andersson	

### Other articles

<b>Euratom On-Site Laboratories Refurbishments and Developments .....</b>	<b>43</b>
L. Duinslaeger, K. Casteleyn, C.-I. Andor, R. Buda, R. C. Marquez, L. Commin, F. D'Amati, L. Emblico, T. Enright, M. Hild, J. H. Domenech, A. Le Terrier, K. Luetzenkirchen, S. Mialle, S. Morel, A. Muehleisen, A. Nicholl, C. Nourry, D. Pouban, M. Ramos Pascual, G. Rasmussen, A. M. Sanchez Hernandez, F. Sarli, S. Street, G. Suurkuusk, G. Toma, M. Toma, O. S. Valu, P. Van Belle, D. Wojnowski, E. Zuleger, F. Lipcsei, P. Schwalbach, S. Synetos	

## Editorial

O. Okko

I am pleased to provide you the 56th edition of the ESARDA Bulletin, containing papers related to encapsulation plants and geological repositories presented at the ESARDA Working Group meetings in Ispra in November 2017. Some of the papers were already selected by the chairpersons of the 49th ESARDA Symposium, held in Dusseldorf on May 2017, that, for editorial reasons, were not published in earlier issues. As discussed with the Editorial Committee, the aim was to produce a thematic issue on the different aspects related to the geologic disposal of spent nuclear fuel. This is also a reflection to the feedback of the member survey carried out during the winter 2017-2018 by the ESARDA Reflection Group.

As the past Chair of the Verification Technologies and Methodologies (VTM) Working Group (WG) I have been pleased to experience interactive participation in the joint meetings with the Implementation Safeguards (IS), Containment and Surveillance (C/S) and Non-destructive Assay (NDA) WGs during the last years in Ispra November meetings. The non-destructive verification of spent nuclear fuel prior to the encapsulation has been one of the main topics of the NDA WG. I am grateful to the presenters for their work to produce scientific articles for the ESARDA Bulletin. All articles were reviewed by at least two independent expert reviewers, guaranteeing the high standard of the publication.

On behalf of ESARDA, I would like to raise the status of our Bulletin. The periodic issuance depends on the members'

activeness. The symposium papers give a basis for new articles, but in order to catch all advances in our multi-disciplinary field of safeguards, I encourage you to submit contributions to the Bulletin at any time so that all the ESARDA community can benefit from the latest progress and achievements in safeguards research and development. Authors are kindly requested to follow the formatting instructions available on the ESARDA website.

I am very pleased to remind you of the following ESARDA events coming in 2018-2019:

- Some of the Working Group meetings will be arranged already in connection with the IAEA Safeguards Symposium in Vienna in early November 2018.
- ESARDA will celebrate its 50 years anniversary during the ESARDA Symposium in Stresa (Italy) on 13–16 May 2019.

Detailed information will be posted on the ESARDA website.

Regarding the ESARDA website and the new LinkedIn ESARDA profile, I would like to address sincere thanks to Andrea De Luca, webmaster and essential assistant for the ESARDA Bulletin preparation, and to Elena Stringa, the Bulletin Editor, for arranging the timely review process. Thank you very much for your engagement during the preparation of this thematic issue.

I wish you all a successful, fruitful and relaxing summer 2018.

# GOSSER - Geological Safeguards and Security R&D Project in Finland - How STUK prepares itself for the Final Disposal in Finland

O. Okko, M. Moring, T. Honkamaa, E. Martikka, M. Hämäläinen

Radiation and Nuclear Safety Authority  
STUK, Finland

*The licensed construction of the disposal facility in Finland began in autumn 2016 as the foundation works for the encapsulation plant and the excavations of the access tunnels to the canister shaft and canister storages in the geological repository were initiated. The disposal of spent nuclear fuel is scheduled to start in Finland in mid 2020's after the operational licence is granted. To ascertain that necessary technical safeguards tools are available at that time, STUK, the Radiation and Nuclear Safety Authority of Finland has set up a national R&D project GOSSER (Geological Disposal Safeguards and Security R&D). GOSSER's main objective is the finalisation of the national Finnish concept for safeguarding the final disposal of the spent nuclear fuel. This concept and related R&D efforts are coordinated with the Finnish facilities, European Commission and the IAEA. Activities in GOSSER include so far: 1) Participation in R&D of robust, reliable, and accurate methods to verify spent nuclear fuel prior to final disposal. The work has been done in cooperation with Helsinki Institute of Physics, the IAEA and other international partners. 2) Participation in the Safeguards-by-Design process of the Finnish encapsulation plant and final repository and, when necessary, development of safeguards methodologies for attaining knowledge of the verified nuclear material and to maintain it for future generations.*

**Keywords:** Spent Nuclear Fuel; Safeguards; IAEA; Geological Disposal; Final Disposal; Safeguards-by-Design

## 1. Introduction

In November 2015 the Finnish Government granted the licence to construct the disposal facility consisting of the encapsulation plant (EP) and the geological repository (GR). The operator (Posiva), Finnish State Regulatory Authority (STUK), IAEA and the European Commission are cooperating on developing safeguards measures and on designing the necessary safeguards infrastructure for these facilities. The spent fuel disposed of will not be accessible for verification using traditional safeguards measures. The international and national safeguards measures have to create confidence that no nuclear material is diverted before, during or after the disposal process and that no undeclared nuclear activities take place at the disposal facilities. Moreover, the operational phase of the facilities will last over a century, thus the safeguards-related technological infrastructure should be

flexible and upgradable. Safeguards by design (SbD) e.g. planning the safeguards measures and designing the necessary safeguards infrastructure during the design phase of the facilities has many benefits. Cost-efficiency is assured by including safeguards equipment such as cameras, radiation detectors, cables and conduits, into the facility design.

A plan for the operator's safeguards activities during the construction and operation of the disposal facility was included by the operator in the application for the construction licence. This included the main steps in nuclear material accountancy and control during the facility development and preliminary plans for the control and accountancy during spent fuel transfers through the encapsulation and disposal process. The plan was approved by STUK during the licensing process and an assessment was included in the STUK Statement [1]. However, in order to ascertain that necessary technical safeguards tools are available at the time needed, STUK launched the national R&D project GOSSER (Geological Disposal Safeguards and Security R&D). The main objective of GOSSER is the finalisation of the national Finnish concept for safeguarding the final disposal of the spent nuclear fuel. This concept and related R&D efforts are coordinated with the Finnish operators, the European Commission and the IAEA.

The key task of GOSSER (named LOVE) is to develop a robust, reliable, and accurate method to verify spent nuclear fuel prior to final disposal. The IAEA requires that spent fuel is verified at a partial defect level before transfer to "difficult to access" locations; however, there is no current method available that can reliably detect a diversion of less than 50% of the pins in a fuel element. The Finnish Support Programme to the IAEA Safeguards has researched the applicability of Passive Gamma Emission Tomography (PGET), and it will be the main candidate for further investigation. Combined with other methods, like gamma spectrometry and neutron measurements, it can be used to verify the correctness and completeness of the declared fuel at pin level. Another task of GOSSER (named JOY) is to evaluate and, when necessary, develop safeguards methodologies for attaining knowledge of the verified nuclear material and to maintain it for future generations. This task may require different techniques from traditional C/S, including geophysics and novel technologies, as well as methods from societal



verification and long term data management. GOSSER will recognise the interfaces between safeguards, security and safety [2]. Security and safeguards both share a common objective: spent nuclear fuel is secured from unlawful actions.

## 2. Verification of spent fuel prior to disposal

STUK has a regular NDA verification programme. The goal of this programme is to verify that information provided by the operator is correct and complete, maintain and develop NDA expertise, prepare for final disposal and support IAEA safeguards conclusions. STUK performs 1 – 2 measurement campaigns annually at each Finnish NPP site Olkiluoto and Loviisa. The traditionally used verification tools are SFAT, eFORK and GBUV [3]. Since early 2017 year also the PGET device is used for verification as well as testing [4].

The Finnish Support Programme to the IAEA Safeguards has studied the applicability of Passive Gamma Emission Tomography (PGET) [5]. Under the GOSSER project, a research group was established in 2015 to study and develop the PGET method further. The Finnish Funding Agency for Innovation (TEKES) provides funding for the Finland Distinguished Professor Programme (FiDiPro) at the Helsinki Institute of Physics (HIP) for the years 2015 – 2018. STUK has a guiding role in the work and also actively participates in method development. The aim is to develop a combination of robust, reliable, and accurate methods to verify spent nuclear fuel prior to final disposal, down to detecting diversion of single fuel pins. Because the IAEA and GOSSER project share the same main goal, to develop functional apparatus for partial defect level spent fuel verification, the LOVE project can provide in-kind support to the work conducted under IAEA MSSP tasks. This will include, for instance, arranging test campaigns with the NPPs.

The latest tests with the prototype have shown the applicability of the method. Combined with other methods, like gamma spectroscopy and neutron measurements, it can provide precise and accurate verification results. The first campaigns with the upgraded PGET in Finland took place in February 2017 in Loviisa and in April in Olkiluoto. The campaigns went very well. The deployment of the system was easy (see Fig. 1) and the PGET demonstrated its ability to reconstruct and analyse images of various fuel types with relatively short acquisition times (about 5 min). Missing pins were detected with good confidence. Examples from these measurements are presented in Figure 2. The progress is reported also in the MSSP task report [6]. It is also foreseen that the PGET will be authorised for spent fuel verification measurements. Although the technology has been developed and demonstrated, some research is still needed to support system development.

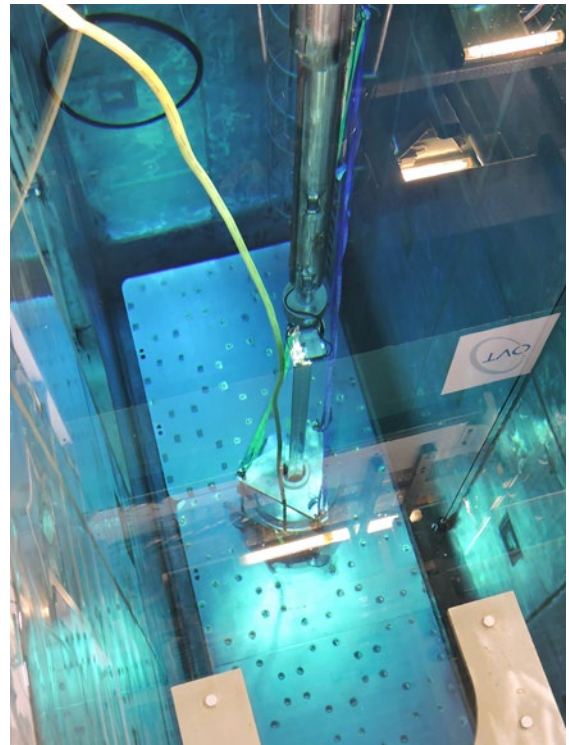


Figure 1. The PGET in spent fuel pool.

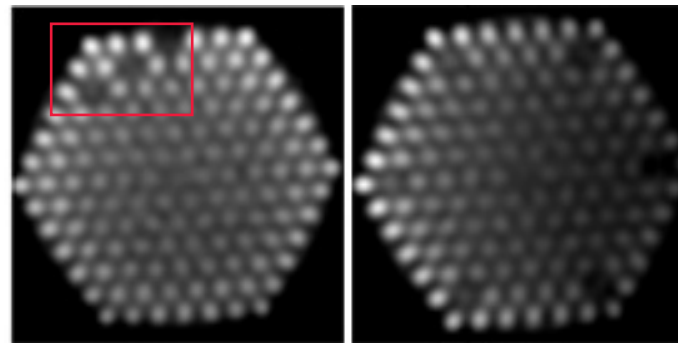
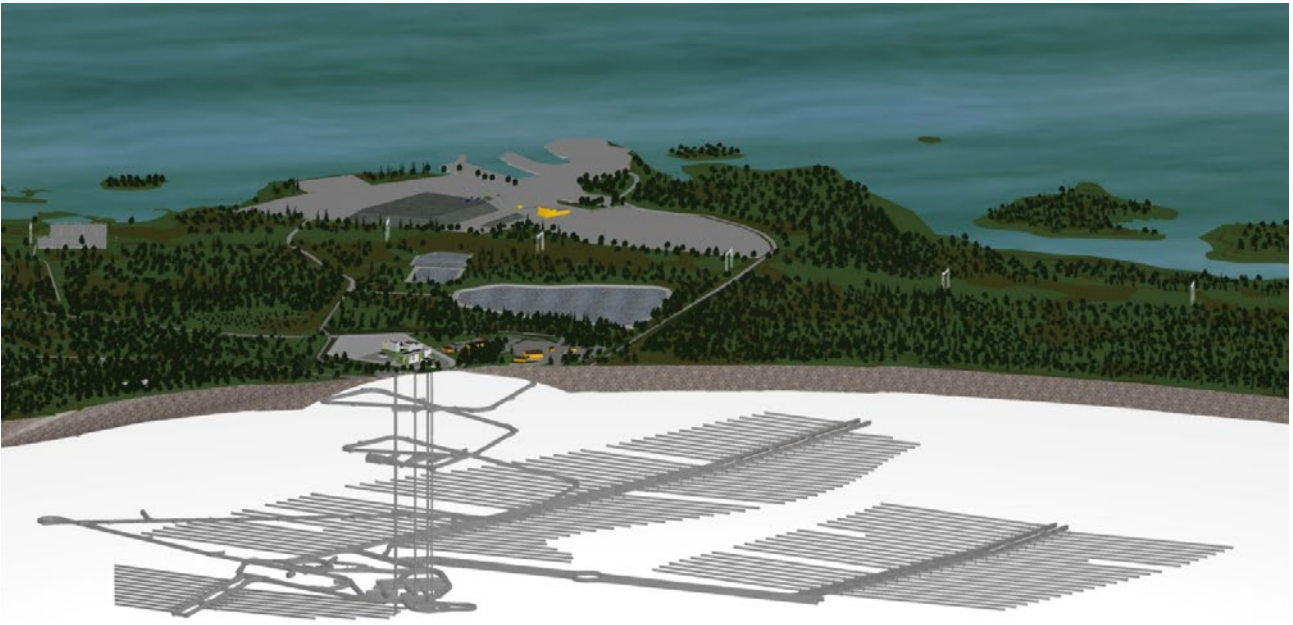


Figure 2. Examples from PGET measurements [5].

## 3. Safeguards-by-Design process

In addition to the NDA measurements several other safeguards practices and measures are to be developed and implemented with the facility design, construction and commissioning. The safeguards equipment infrastructure to be installed in the Olkiluoto encapsulation plant is already developed in cooperation between the stakeholders, IAEA, European Commission, STUK and the operator [7]. However, the design of the facility is still being optimised by the operator. Continuous communication between the stakeholders is essential to assure that the operator maintains safeguardability of the facility and that the inspectors are able to modify their equipment infrastructure according to changes in plant design. A similar process is foreseen to be conducted for geological repository during the initial planning and construction phase. Geological investigations and construction of the geological repository will continue in parallel through its operational period. Due



**Figure 3.** Layout of the disposal facility at Olkiluoto.

to unforeseen elements in the geology and rock mechanics, the repository layout at Olkiluoto cannot be rigidly planned in advance, so any safeguards measures in the repository needs to have enough flexibility to adapt to design changes. The current layout of the facility is shown in Figure 3. The basic technical characteristics (BTC) and the site declaration are to be updated when major changes are introduced in the design process.

The operator presented its plan to control the integrity of the fuel canisters and to demonstrate and to document their safe transfer to the emplacement hole with their construction licence application. This plan was approved by STUK in 2015 with the remark that the operator has to facilitate safeguards measures by STUK, the EC and the IAEA with the progress of the project. Currently, the material accountancy for fuel canisters is a part of the negotiations of the Facility Attachment to be agreed between the IAEA and the EC. Both STUK and the operator are consulted from the beginning of this iterative process. Also, the ultrasonic identification of the canisters is under method development [8]. In the disposal process, the Continuity-of-Knowledge and supporting Containment and Surveillance measures will be essential; whereas the annual physical inventory verification (PIV) of the underground repository cannot be carried out in a traditional manner. The Safeguards-by-Design process will cover also these aspects.

In order to detect undeclared activities, STUK has direct access to and, in cooperation with safety, also de facto full time institutional presence at the active final disposal facility site. In the national concept development this asset will be utilised. Currently the Olkiluoto monitoring programme

was reassessed in [9]. and a few recommendations were suggested to have more safeguards use of the operators data and safety assessment. STUK follows the daily research and work plans, and the continuous monitoring of the site as safety assessment of a geological repository is of international research interest with also societal and safeguards aspects e.g. [10]. STUK has also contacts to other authorities in Finland that are e.g. licensing construction activities and therefore can report about any undeclared safeguards-relevant activities. In contrast to this, the international safeguards inspectorates lack these capabilities, therefore, they have to employ technological solutions, which STUK has less need for that. STUK however must be aware of the capabilities and properties of these techniques. As the GOSSER project is based on external cooperation, STUK does not need to perform its own research for this purpose. It is sufficient to follow what other institutions are developing in Finland and abroad for the safety assessment and security precautions and to demonstrate this to the inspectorates.

#### 4. Summary

The GOSSER project was launched because safeguards for spent fuel disposal is a new challenge and new concepts need to be developed and implemented already during the early design and construction of the final disposal facility licensed in 2015. The time span of the overall disposal project is more than 100 years so process optimisation has high pay off opportunities. As the disposal facility is of a new kind to be safeguarded, the methods developed and applied in Finland have to gain international acceptance.

The disposal of spent fuel requires that safety, information security and other security arrangements and the safeguards required to prevent the proliferation of nuclear weapons are properly implemented. This requires the reconciliation of all areas resulting in the implementation of 3S in an appropriate manner. This, in turn, requires action from the operators producing, encapsulating or disposing of spent nuclear fuel as well as the authorities (STUK). Although, the European Commission and the International Atomic Energy Agency (IAEA) have strong roles in the safeguarding of nuclear materials they are not directly in the focus of this project, however, they will benefit from its developments.

The novelty of the disposal concept calls for adequate research and provides the reasoning for establishment of GOSSER R&D project. If GOSSER is not successful, in the worst case there is a risk that the credibility of the disposal concept is questioned and; moreover, the future generations may not have adequate information to satisfy themselves that the spent fuel is fully and reliably disposed of in the repository. The main objective of GOSSER is the finalisation of the Finnish concept for safeguarding the disposal of the spent nuclear fuel by 2018.

## 5. References

- [1] STUK, *STUK's statement and safety assessment on the construction of the Olkiluoto encapsulation plant and disposal facility for spent nuclear fuel*. STUK-B 196, November 2015.
- [2] Honkamaa, T., Martikka, E. Moring, M. Hack, T., Okko, O. & Hämäläinen, M. *GOSSER - Geological Disposal Safeguards and Security R&D Project in Finland - How We Prepare Ourselves for 2023 When Final Disposal Starts in Finland*. INMM 57th Annual Meeting July 24 – 28, 2016, Atlanta, Georgia, USA.
- [3] Honkamaa T. & Hämäläinen M. *How STUK verifies Spent Nuclear Fuel and Why?* in Proceedings. ESARDA 25. Annual Meeting. Symposium on Safeguards and Nuclear Materials Management, Stockholm, 2003.
- [4] Tobin, S. J., Peura, P., Bélanger-Champagne, C., Moring, M., P. Dendooven, P. & Honkamaa. Utility of including passive neutron Albedo Reactivity in an integrated NDA System for Encapsulation Safeguards, ESARDA Bulletin 56, p. 12-18, 2018.
- [5] IAEA 2017. *Technologies Potentially Useful for Safeguarding Geological Repositories*. ASTOR Group Report 2011 – 2016, STR-384, 107 p.
- [6] Honkamaa, T., Levai, F., Turunen, A., Berndt, R., Vaccaro S. & Schwalbach, P. *A Prototype for passive gamma emission tomography (PGET)*, in IAEA Symposium on International Safeguards, 20 – 24 October 2014, Book of Abstracts, IAEA CN-220-189, p. 257.
- [7] Ingegneri M., Baird K., Park W.-S., Coyne J. M., Enkhjin L., Chew L. S., Plenteda R., Sprinkle J., Yudin Y., Ciuculescu C., Koutsoyannopoulos C., Murtezi M., Schwalbach P., Vaccaro S., Pekkarinen J., Thomas M, Zein A., Honkamaa T., Hämäläinen M. , Martikka E., Moring M., Okko O. *Safeguards by Design at the Encapsulation Plant in Finland*. IAEA Symposium on International Safeguards, 20 – 24 October 2014, Book of Abstracts, IAEA CN-220-023, p. 207.
- [8] Littmann, F., Clementi, C., Capineri L., Andersson, C. Stål, J.O., *Ultrasonic identification methods of copper canister for final geological repository*, ESARDA Symposium 2017, Jülich/Düsseldorf 15 -18 May 2017.
- [9] Pentti E. & Okko, O. *Usability of Monitoring at the Olkiluoto Repository Site for Safeguards*. ESARDA Bulletin 56, p. 28-35, 2018.
- [10] Okko, O. *Safeguardability of a geological repository for spent nuclear fuel*, pp. 583 - 598 in Apted, M. and Joonhong, A. (Eds.) *Geological Repository Systems for Safe Disposal of Spent Nuclear Fuels and Radioactive Waste 2nd Edition*, Elsevier 2017.



# Non-destructive assay sampling of nuclear fuel before encapsulation

C. Hellesen and S. Grape

Division of Applied Nuclear Physics, Department of Physics and Astronomy, Uppsala University, Box 516, 751 20 Uppsala, Sweden  
E-mail: carl.hellesen@physics.uu.se, sophie.grape@physics.uu.se

## Abstract:

*Swedish spent nuclear fuel is planned to be verified before being encapsulated and placed in a geological repository 500 meters below ground in the bedrock. Verification before encapsulation is intended to ensure both the safe storage of the spent fuel, and that Sweden is honoring its international obligations according to the NPT, international treaties and bilateral agreements on the topic of nuclear safeguards.*

*The measurements will mark the last chance of verifying the spent fuel, as there are no plans to retrieve them once they enter the geological repository. With respect to nuclear safeguards, fuel assemblies will likely be verified for both gross defects and partial defects, whereby a fraction of the fuel content has been removed or replaced. A conclusion also needs to be made on the correctness of the fuel assembly declarations, translating into a verification (or determination) of the fuel parameters such as initial enrichment, burnup and cooling time.*

**Keywords:** non-destructive assay; spent nuclear fuel; sampling; encapsulation

## 1. Introduction

Sweden is a country with a substantial fraction of electricity being produced by nuclear power. The spent nuclear fuel is currently stored in the Swedish Interim Storage Facility for Spent Nuclear Fuel (Clab) in Oskarshamn. The proposed future encapsulation facility Clink will be built next to Clab, and it is planned that the fuel will be measured here before being encapsulated and shipped to the geological repository in Forsmark. This repository will be a so-called “difficult-to-access storage” since the fuels are not easily retrievable. This means that spent fuel assemblies that can be dismantled should be verified with a partial defect test or, if not available, the best available method approved for inspection use; spent fuel assemblies that are difficult to dismantle (e.g., welded fuel assemblies) should be verified with at least a gross defect test. The reason is that spent nuclear fuel contains up to 1% of plutonium, which is a nuclear-weapons usable material.

There is a need to develop a sampling plan for the spent nuclear fuels. This sampling plan needs to include sufficiently many fuels in the fuel inventory in a partial defect test to ensure that the probability to discover a missing

specified quantity is at least 90% or greater [1]. Often, a specified quantity is equal to a significant quantity (SQ), which for plutonium is 8 kg [2]. This means that in total, the false-negative rate of a diversion of 1 SQ should be below 10%.

The measurements will be non-destructive, possibly including several measurement techniques in one location. According to the IAEA, the measurements will take place in “the assembly handling cell” [3] before placing the fuels in the copper canister. It is not clear at this stage what this location corresponds to; possibly it could mean the fuel handling pool at the upcoming Clink facility, but not necessarily. If we assume that the measurements will take place under wet conditions, measurements could be performed on the pool side, in a dedicated measurement room next to the pool via collimators, inside the pool or at fuel racks or integrated into the fuel handling equipment [4].

Although this context is inspired by Sweden, the situation is similar to that of many other countries. The IAEA requirements are the same everywhere, and it is likely that the implementation of nuclear safeguards verification in a pioneering country becomes the standard, or at least a guide, for other countries.

## 2. Experimental verification challenges before encapsulation

There are several open questions remaining with respect to the nuclear safeguards verification, and specifically with relevance to the sampling plan. Having said that, we here consider that not all fuels will be measured with the same accuracy (given that instrument accuracies are expected to improve over time), and that some kind of random sampling will be performed. Examples of relevant questions in this context are:

- What exactly needs to be measured?
- What level of defects need to be verified with which confidence?
- What accuracies and uncertainties are associated with the considered measurement techniques or instruments?

A spent nuclear fuel typically contains up to 1% plutonium, which is particularly sensitive as it can be used as raw material in a nuclear explosive device (NED). It is classified by IAEA as a direct use material [2]. This is in contrast to the



uranium in the fuel, which is low enriched (LEU) and cannot be used in a NED without subsequent enrichment. LEU, like irradiated nuclear fuel, is classified by IAEA as indirect use nuclear material. Since plutonium can be chemically separated from other elements in the fuel it is vital that the spent fuel is accounted for.

For all weapons usable materials, a significant quantity (SQ) is defined as the approximate mass for which the construction of a NED cannot be excluded. For plutonium this is 8 kg [2]. This means that a PWR assembly, with a weight of about 600 kg, contains up to 6 kg plutonium. If a larger fraction than  $\frac{2}{3}$  of the fuel rods in two separate assemblies are diverted and replaced with dummy rods of equal mass, it could make up one SQ of plutonium. Furthermore, if smaller fractions are diverted from a, potentially large, number of fuel assemblies, one significant quantity of plutonium can be accumulated over time, a scenario known as roll-up [5]. To exclude the risk of a roll-up scenario in an encapsulation plant, where on the order of 100 000 assemblies will be encapsulated, will obviously be a challenge. It is important that the fuel assemblies are tested for defects at different levels prior to encapsulation. In this paper we consider diversions from 50% of an assembly down to single pin level.

Within nuclear safeguards, it is important to assure that a significant quantity of weapons usable material is not diverted. The diversion can take many shapes. If we consider 17 by 17 PWR assemblies, each rod contains about 20 grams of plutonium. Acquiring a total of one SQ would require the removal of one single pin from 400 PWR different assemblies. It can be argued that it is sufficient for nuclear safeguards inspectors to detect one single diversion attempt *before* the series of diversions have accumulated to a total of one SQ, rather than necessarily detecting 400 cases of partial defects on the single rod level. A positive outcome in a random sample could then initiate sampling on a more detailed level of a larger number of assemblies to investigate the possibility of an actual diversion attempt (as opposed to a mistake, such as a wrongly declared assembly).

A sampling protocol would have to be established. Course measurement techniques, capable of detecting diversions involving a large fraction of missing fuel pins, would need to be used more frequently. Very accurate techniques, capable of detecting diversions of small fractions of missing pins, could be used less frequently.

There are a limited number of instruments currently authorized for partial defect testing on the 50% level by the IAEA: the Digital Cherenkov Viewing Device (DCVD) [6] and the Fork detector (FDET). In addition, the passive gamma emission tomography (PGET) system [7] has also recently been authorized for genuine partial defect verification, meaning verification on the single pin level. Work on

improving these techniques, as well as the development of new ones, is ongoing. See for example [8].

### 3. Methodology

In this work we have studied the defect verification of a fuel inventory intended for encapsulation with a Monte Carlo based sampling approach. Two types of simulations have been used.

In the first type, the sampling is tested at three levels of defects, 50%, 10% and single pin. The defect level is here referred to as  $f_{\text{defect}}$ . In the first case, 50% of the rods, in a randomly selected small subset of the assemblies, have been substituted with dummy rods of the same mass. In the second case, 10% of the rods have been substituted, and in the third case, single rods are substituted.

The 50% defect corresponds roughly to the detection capability of the digital Cherenkov viewing device (DCVD) [6] and the single pin defects correspond to the detection capability of the passive gamma emission tomography (PGET) [7]. Currently, there is no authorized technique for detecting defects at the 10% level. However, one objective of this study is to evaluate the role for such a technique, and we refer to it here as the 10%-technique.

The diversion scenario considered in this study is a continuous accumulation of smaller quantities of fissile material, also referred to as roll-up. To our knowledge this is the first technical study that investigates detection level as a function of diversion scenario and instrument capability in connection to verification of a large inventory of spent nuclear fuel.

When evaluating the effectiveness of different fuel verification techniques, two parameters are of central importance: the true positive rate as well as the false positive rate; these are denoted here as  $p_{\text{detect}}$  and  $p_{\text{false}}$ . The value of  $p_{\text{detect}}$  is the probability that a fuel assembly with a partial defect is correctly identified as such. On the other hand,  $p_{\text{false}}$  is the probability that an untampered fuel assembly is mistaken for a partial defect assembly. The two parameters play different roles, which is further discussed in section 6. These probabilities are connected to the measurement techniques used and are given as input to the Monte Carlo sampling. The probability of a successful cumulative diversion of, in total, one SQ without detection is denoted as  $P_{\text{divert}}$  and is a result of the simulation.

In the simulation, a random sampling of the fuel inventory is made.

- First, a number of assemblies in the inventory are selected randomly for diversion, in total adding up to SQ of plutonium.
- Second, a number of assemblies in the inventory are selected randomly for verification. The fraction of verified assemblies is referred to as  $F_{\text{sample}}$ .

- Third, for each verification, randomized defect tests are generated based on the probability  $p_{\text{detect}}$ . If a test scores a true positive, the diversion attempt is considered to have failed.

In the first simulation type,  $F_{\text{sample}}$  is varied from  $10^{-3}$  to 1.0, and different levels of  $p_{\text{detect}}$  are tested. With this procedure we evaluate the required sampling frequency at different levels of  $p_{\text{detect}}$  for diversions at fixed defect levels (50%, 10% and pin level).

However, the detection probability of an instrument is typically not a constant but will vary with the defect level. For this reason, we also perform a second type of simulation with diversion attempts at different levels of defects, starting from single pins and up to 50% defects. Further, in the second simulation  $p_{\text{detect}}$  is not kept constant. For the DCVD and the 10%-technique, we assume that  $p_{\text{detect}}$  is a function that increases with the size of the defects; while for the PGET we assume that  $p_{\text{detect}}$  is close to 1.0 for all defect sizes.

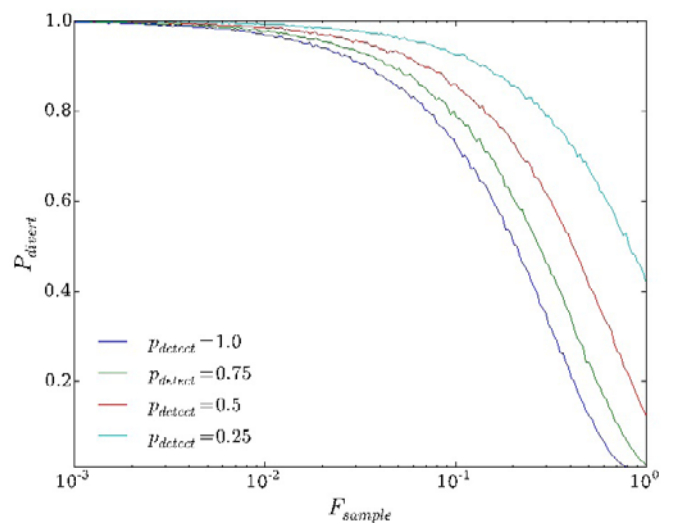
In [9] the efficiency of a DCVD for partial defect detection was investigated. It was concluded that the measured light intensity from spent fuel assemblies agreed with the modeled intensity within  $\pm 30\%$ . If the level of  $p_{\text{false}}$  should be kept acceptably low, a threshold for defect detection can be used that corresponds to  $p_{\text{detect}} = 1.0$  around a 50% defect level. For defects at a 25% level,  $p_{\text{detect}}$  will drop to 0.5, and for defects close to the single pin level  $p_{\text{detect}}$  will be 0. In the modeling we therefore assume that  $p_{\text{detect}}$  follows a soft step function through these points. Similarly, we assume that  $p_{\text{detect}}$  for the 10%-technique follows a similar soft step function but reaches 1.0 for 10% defects.

The procedure in the second simulation type is similar to what is described above for the first type, with the difference that all three techniques are used together, but with different sampling frequencies. Further, if a positive result is found, the same assembly is always tested again with a more accurate technique. For example, if the DCVD makes a positive measurement, it is re-verified with the 10%-technique, and if the 10%-technique makes a positive measurement it is re-verified with the PGET. Once the PGET makes a true positive measurement, the diversion attempt is considered to have failed.

In the Monte Carlo simulation of the sampling we have here assumed that all assemblies sent for encapsulation are 17 by 17 PWR type with 264 fuel pins and contain 1% plutonium. This is a simplification, but these assumptions are trivial to change to the exact conditions for the SNF inventory under consideration. For example, the Swedish inventory consists of a mix of PWR and BWR fuels with burn up levels varying from about 10 to 60.

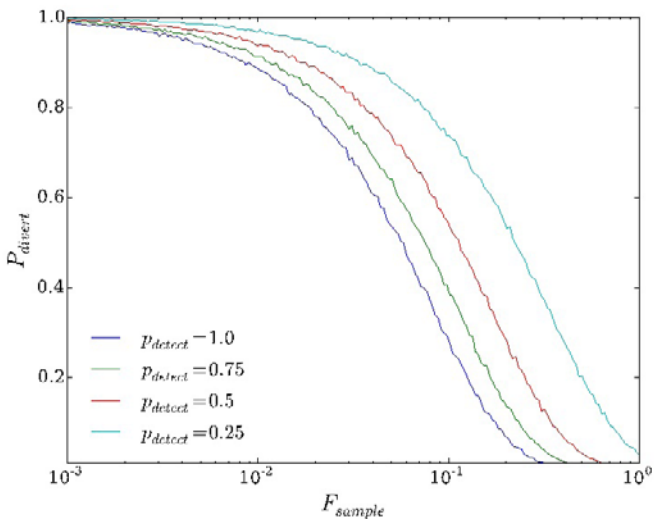
## 4. Results

The results from the first simulation type are presented in figures 1 through 3. In figure 1 we show the probability for diversion success of one SQ plutonium ( $P_{\text{divert}}$ ) as a function of sampling frequency. In this case the partial defects are on the level of 50% in 17x17 PWR assemblies. The results show that, given a 95% detection probability of diversion attempts ( $P_{\text{divert}}=0.05$ ) and  $p_{\text{detect}}=0.5-0.75$ , every fuel assembly needs to be verified in order to ensure that one SQ of plutonium has not been diverted. Should the sampling be made less frequently, or with a less accurate method, it cannot be ruled out that a diversion of one SQ has been made at some place in the fuel inventory.



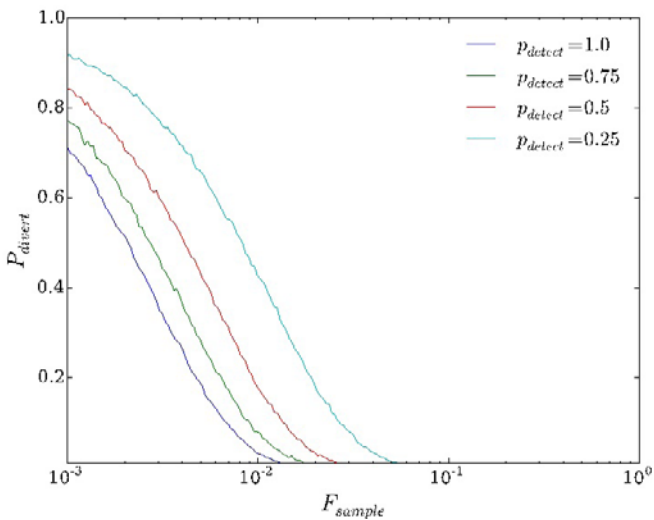
**Figure 1.** Probability for diversion success of in total one SQ if fuels suffer from a 50% partial defect level. The different curves correspond to different values of  $p_{\text{detect}}$ , indicating the level of accuracy of the selected instrument at this level of partial defects.

Continuing with figure 2, we show the results of the sampling for the case of defects at a 10% level. Here, the sampling frequency and accuracy needed to detect a total diversion of one SQ are inversely related. If the accuracy ( $p_{\text{detect}}$ ) is close to 1.0, i.e. the risk of false negatives is small; it suffices to test for diversions at this level with a sampling frequency of around 1/4. However, if the accuracy of the chosen method is lower, the sampling must be done more frequently (approaching every assembly) in order to achieve a reasonably low risk for a successful diversion at some place in the inventory.



**Figure 2.** Probability for diversion success of in total 1 SQ if fuels suffer from a 10% partial defect level. The different curves correspond to different values of  $p_{detect}$ , indicating the level of accuracy of the selected instrument at this level of partial defects.

In figure 3 the situation for partial defects at the single pin level is shown. Qualitatively it is similar to the situation described above, although the required sampling frequencies are more than one order of magnitude lower.

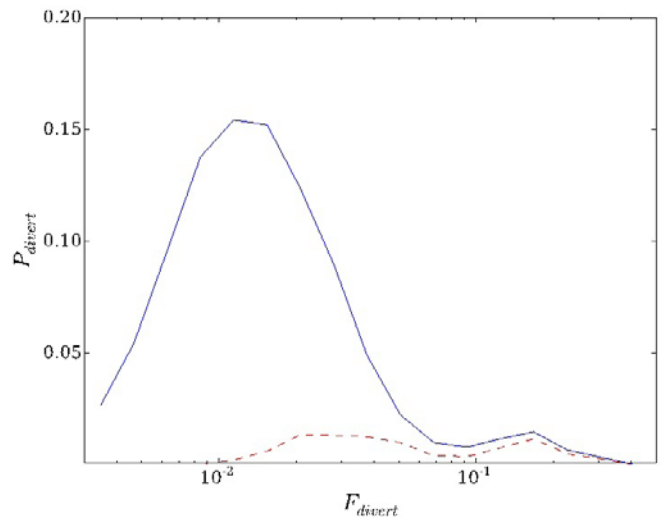


**Figure 3.** Probability for diversion success of in total 1 SQ if fuels suffer from a single rod partial defect level. The different curves correspond to different values of  $p_{detect}$ , indicating the level of accuracy of the selected instrument at this level of partial defects.

Finally, in figure 4 we show the results of the second simulation type with all three instruments operating in parallel. The different sampling rates used are based on the results presented above, from the first simulation type. For the DCVD and the 10%-technique, we sample every assembly and every 1/4 assembly, respectively. However, for the PGET we test two sample rates. The two curves in figure 4

correspond to sampling every 1/100 assembly with the PGET, while the dashed red curve corresponds to sampling every 1/20 assembly.

While a sampling of every 1/100 assembly with the PGET is adequate to detect a diversion of 1 SQ from multiple single pin defects, there is a blind spot for diversions around 2%, which corresponds to about 5 pins per assembly. For such diversions, 1 SQ is acquired too quickly, and the 10%-technique is not yet sensitive enough to detect the defects. However, with a sampling of every 1/20 assembly with the PGET (dashed red curve) the blind spot is reduced significantly.



**Figure 4.** Probability for successive diversion of in total 1 SQ at varying levels of defects, from single pin to 50%. The solid blue and dashed red curves correspond to sampling 1/100 and 1/20, respectively, of all fuel assemblies with the PGET instrument

Some implications can be noted from the results in this paper. Diversions can take place with varying levels of defects, and the sampling procedure must take this into account. If a diversion scenario considers large defects from a few assemblies, one SQ can be acquired relatively fast. Consequently, a sampling of every assembly has to be made with a quick and robust method that can reliably detect large defect levels. Here we used a maximum of 50% of the rods missing, and the probability for true positive detections,  $p_{detect}$ , must be kept around 75% or higher. On the other hand, if a diversion is made from small quantities over long times, it is only necessary to sample a (small) subset of the assemblies to detect one manipulated fuel assembly. However, the measurement technique must be capable of detecting small defects. We used an example of partial defects on the single rod level, and depending on the level of  $p_{detect}$  somewhere between 1/500 and 1/100 of all assemblies must be assayed. However, this number increases to about 1/20 of all assemblies when



## 5. Discussion

The nuclear safeguards verification needs to be non-intrusive and interfere with regular operations of the facility to a minimum extent. Currently, the DCVD [6] is authorized to be used for verification of defects on a level of 50%. The operation of a DCVD is straightforward, quick and in many aspects non-intrusive, and sampling every assembly should not pose a serious interference with the routine operations of the facility.

At the other end of the spectrum, passive gamma emission tomography (PGET) of the fuel assemblies has the potential to detect single missing rods from fuel assemblies. In comparison to the DCVD, it is more time consuming and requires the fuel to be moved to a dedicated measurement station. Sampling every assembly would likely result in a serious interference with the operations of the facility. However, individual diversions at the level of a few pins are small enough to allow for a sampling frequency at 1/20 or lower, which could be more easily incorporated into the facility operations. Under the assumption that one copper canister consisting of four PWR assemblies is filled each day, it would only be necessary to use the PGET a few times every week.

However, with the currently available measurement techniques, detecting possible diversions at a defect level of about 10% poses a considerable challenge. Actually, this concerns any partial defect level between 50% and a few pins. Such defects cannot be reliably detected with the DCVD technique and there is no other technique existing today for partial defect verification at that level today. While a PGET device could in principle be used to detect defects at about 10% with very high accuracy ( $p_{\text{detect}} \approx 1.0$ ), the required sampling frequency would still be comparably high. In the case of a facility for encapsulation of PWR assemblies, around 1/4 of the assemblies would have to be sampled. This would mean that the PGET would have to be operated several times per day.

Instead, adding a third measurement technique capable of reliably detecting defects at the 10% level, but with a measurement time significantly shorter than the PGET, is preferable. With three systems running in parallel, a robust verification of defects at all levels can be made with little interference with the routine operations of the facility. The DCVD could be used for all assemblies, the 10%-technique a few times per day, and the PGET a few times per week.

Finally, the importance of  $p_{\text{false}}$ , i.e. the probability for a false positive result, must also be considered. A likely sampling procedure would be that a positive result using a course technique, e.g. a DCVD scanning for 50% defects, is followed by an examination using a more accurate technique designed for defects at the 10% level in order to verify if the positive result is an actual defect or a measurement

error. Likewise, a positive assay with a technique designed to detect defects at the 10% level would be followed by an examination using the most precise instrument, the PGET.

However, if  $p_{\text{false}}$  is too high, the use of the more accurate measurement technique would be dominated by verifying false positive assemblies identified by less accurate techniques. Optimizing the sampling procedure requires detailed knowledge of  $p_{\text{detect}}$  and  $p_{\text{false}}$ , which is beyond the scope of the paper. But we do note that the two probabilities are likely related. Setting a low threshold for a positive result, can increase the value of  $p_{\text{detect}}$ , which would result in a lower required sampling frequency. But at the same time, a lower threshold is also likely to result in a higher false positive rate, which would increase the usage of more precise instruments. Likewise, while longer measurement times interfere more with the operations of the facility, they can also potentially reduce noise and increase  $p_{\text{detect}}$  as well as decrease  $p_{\text{false}}$ , and therefore lessen the interference with the facility operations.

## 6. Conclusion

This simulation study aims at investigating how spent nuclear fuel can be sampled for the purpose of verifying defects, for instance before encapsulation and placement in a difficult-to-access storage. The fuel assemblies are expected to be measured in order to draw conclusions on gross and partial defect verification. The nuclear safeguards verification needs to be non-intrusive and to interfere with regular operations of the facility to a minimum extent.

The results show the importance of having an available selection of partial instruments capable of detecting varying levels of partial defects. Instruments that can quickly survey a large fuel inventory and give results on whether or not a large fraction of the fuel material has been diverted, are valuable in excluding a diversion scenario where large amounts of nuclear material are removed from a few items. On the other hand, instruments that are able to perform partial defect verification on a low level, e.g. single rods missing, are very valuable for excluding a roll-up scenario.

Currently, there are authorized instruments that can be used to verify partial defects on a level of 50% as well as single missing rods. The results here show that, assuming an interest for keeping the sampling frequency with the PGET low, there is a motivation for developing instruments capable of verifying defects at a level around 10%. Without such instruments, one is referred to an extensive use of the PGET. If a 10%-technique can be found that is quick and robust, its extensive use may not be a problem.

Further, one should keep in mind that partial defect tests might not be the only verification that needs to be made. There could also be an interest to perform additional

assays to verify the fuel parameters (cooling time, burnup and initial enrichment). If such a measurement can provide a defect test at a 10% level as well, it can be used as a complement to a PGET.

Finally, it can also be pointed out that the PGET does not directly probe the fissile content in the assembly, but infers it indirectly from measurements of the gamma emission from fission products. In a diversion scenario where dummy replacement rods have been loaded with Cs-137 a PGET would not detect the diversion.

Consequently, there could also be an interest to employ other methods that directly probe the fissile content of the assemblies, even if such methods have a lower sensitivity than the PGET. Examples are neutron-based techniques such as the differential die away self interrogation (DDSI) [10].

## 7. Acknowledgements

We would like to thank the Swedish Radiation Safety Authority (SSM) for their financial support to this work, under agreements SSM2016-661 and SSM2016-4600.

## 8. References

- [1] International Atomic Energy Agency; Safeguards manual, part: Safeguards Criteria; SMC 14; Annex 4; IAEA; 2003
- [2] International Atomic Energy Agency; IAEA Safeguards Glossary 2001 Edition; International Nuclear Verification Series No. 3; 2002
- [3] International Atomic Energy Agency; Model Integrated Safeguards Approach for a Spent Fuel Encapsulation Plant; SG-PR-1305; IAEA; 2010.
- [4] S.J. Tobin, P. Jansson; Nondestructive assay options for spent fuel encapsulation; Technical Report TR-13-30; SKB; 2013.
- [5] Nuclear Material Control and Accountability, U.S. Department of Energy, 2005.
- [6] J. Chen et al.; Spent fuel verification using a Digital Cherenkov Viewing Device; In Institute of Nuclear Materials Management 46th annual meeting Portland, Oregon; 2009.
- [7] S. Jacobsson Svård et al.; Tomographic determination of spent fuel assembly pin-wise burnup and cooling time for detection of anomalies, 37th annual ESARDA Symposium [8] S. Vaccaro et al., Advancing the Fork detector for quantitative spent nuclear fuel verification, Nuclear Instruments and Methods in Physics Research Section A, Volume 888, 2018, Pages 202-217
- [9] J.D. Chen et al., Detection of Partial Defects using a Digital Cerenkov Viewing Device, IAEA-CN-184/33
- [10] A. C. Kaplan et al., Determination of Spent Nuclear Fuel Assembly Multiplication with the Differential Die-Away Self-Interrogation Instrument, Nuclear Instruments and Methods A, Volume 575, Pages 20-27, 2014.

# Utility of Including Passive Neutron Albedo Reactivity in an Integrated NDA System for Encapsulation Safeguards

S. J. Tobin<sup>1</sup>, P. Peura<sup>2</sup>, C. Bélanger-Champagne<sup>2</sup>, M. Moring<sup>3</sup>, P. Dendooven<sup>2</sup>, T. Honkamäa<sup>3</sup>

<sup>1</sup>Encapsulation Nondestructive Assay Services, Los Alamos, NM 87544, USA.

<sup>2</sup>Helsinki Institute of Physics, FI-00014 University of Helsinki, Finland.

<sup>3</sup>Radiation and Nuclear Safety Authority - STUK, 00881 Helsinki, Finland.

E-mail: steve.tobin.nm@gmail.com, p.peura@iaea.org

## Abstract:

*In August of 2017, the International Atomic Energy Agency (IAEA) issued the “Application of Safeguards to Geological Repositories (ASTOR) Report on Technologies Potentially Useful for Safeguarding Geological Repositories.” In this IAEA report, the nuclear safeguards experts convened made recommendations on various aspects of encapsulation facility and repository safeguards. Specific to the non-destructive assay (NDA) requirements, the ASTOR experts made six specific recommendations. To satisfy these recommendations, a team working under the direction of the Finnish Radiation and Safety Authority researched the capability of an integrated NDA system that combines the capabilities of a Passive Gamma Emission Tomography (PGET) instrument, a Passive Neutron Albedo Reactivity (PNAR) instrument and a load cell. The current study focuses a conceptual design of the PNAR instrument capable of supporting several of the IAEA recommendations. To enable this research goal, the performance of a PNAR instrument, designed to measure boiling water reactor assemblies, was simulated using fuel with the isotopic content representative of fuel of various initial enrichments, burnups and cooling times. The research results illustrate the capability of the PNAR instrument to fulfil the IAEA recommendations while using robust, relatively simple, hardware.*

**Keywords:** non-destructive assay, encapsulation safeguards, PNAR

## 1. Introduction

In August of 2017 the International Atomic Energy Agency (IAEA) issued the “Application of Safeguards to Geological Repositories (ASTOR) Report on Technologies Potentially Useful for Safeguarding Geological Repositories” [1]. For the formulation of this report, the IAEA convened groups of experts on specific topics to provide recommendation on various aspects of encapsulation facility and repository safeguards; this current study focuses on the Finnish implementation of the ASTOR Group recommendations in the context of a non-destructive assay (NDA) system. The mandate of the ASTOR NDA Focus Group was to improve upon the state-of-the-practice given the extremely difficult to access nature of fuel placed in a deep geological

repository. The NDA system proposed by Finland satisfies all the recommendations set forth by the NDA Focus Group by integrating a Passive Gamma Emission Tomography [1, 2, 3, 4] instrument with a Passive Neutron Albedo Reactivity (PNAR) instrument [1, 5, 6, 7]. The purpose of the current study is to describe how the PNAR instrument helps strengthen the NDA system in the implementation of the IAEA recommendations while using robust, relatively simple, hardware that can measure the assembly multiplication.

## 2. Requirements for the encapsulation NDA system

Below is a list of suggested characteristics for the NDA system of a spent fuel assembly encapsulation facility. The list was created by the NDA Focus Group convened by the IAEA as part of ASTOR.

- a. Capability to detect individual pins, even though it is recognized that pin level detection might not be possible for all assembly fuel types and for all burnup and cooling time scenarios.
- b. Capability to verify that the declared assembly is consistent with measured signatures: Enough information is provided in the declaration of each assembly to predict, within useful limits, some measurable signatures from each assembly. Once predicted, a comparison between expectation and measurement is possible and recommended.
- c. Capability to measure assembly neutron multiplication: The neutron multiplication of an assembly can be measured with the neutron signal. Furthermore, this multiplication can also be calculated from the declaration. Multiplication is singled out in this list for its close connection to the presence of fissile material and because it is a bulk property of the assembly.
- d. Robustness, low maintenance and low false alarm rate must all be properties of the NDA system. The NDA system should not significantly impede facility operation; duplicate systems are recommended in addition to using robust technology.
- e. System should be difficult to trick with pin substitution. As noted in Chapter 1 of [1], “all individual NDA techniques ... can be tricked by a well-designed pin replacement.” Hence, the aggregate NDA system needs



to make well-designed pin replacement extremely difficult to plausibly/usefully perpetrate.

- f. Capability to measure the total weight: A measurement of the assembly weight is considered relatively simple and able to contribute one more constraint a 'would be' proliferator needs to satisfy in designing a diversion scenario.

Fulfilling all the characteristics listed above will require an integrated NDA system. The focus of this paper is to describe how the PNAR instrument contributes to fulfilling characteristics (b) through (e).

For characteristic (a) Passive Gamma Emission Tomography (PGET) instrument is expected to provide pin level detection capability in Finland. Recent research on the NDA system intended for the Finnish encapsulation facilities indicates that detection of every pin in a boiling water reactor (BWR) assembly should be straightforward, while detection of every pin in a VVER-440 assembly, is a topic of ongoing research in Finland and at the IAEA [2, 3].

### 3. Passive neutron albedo reactivity physics

The PNAR NDA technique involves the comparison of the neutron count rate for an object when that object is measured in two different setups. One setup is designed to enhance neutron multiplication while the other setup is designed to suppress it. As implemented for the Finnish BWR fuel, the high multiplying section was produced by the assembly in the water of the pool, while the low multiplying section was created by putting 1 mm of Cd as close as possible to the fuel while it remained in the pool. Cd was selected due to its extremely large absorption cross-section for all neutron energies below  $\sim 0.5$  eV. The PNAR signature is calculated by dividing the count rate measured in the high multiplying section by the count rate measured in the low multiplying section.

The first PNAR experiments with an assembly geometry were performed with a 15x15 fresh assembly; a  $^{252}\text{Cf}$  source was imbedded in the assembly to increase the neutron flux. The results showed a healthy change in the PNAR signature with changes in the average initial enrichment of the assembly [5]. An experiment using fresh rods in air was performed inside of a multiplicity counter showing that the sensitivity of a PNAR instrument will increase if the detectors efficiency is elevated enough to support correlated neutron detection [8]. The first use of a PNAR with spent fuel was performed with Fugen fuel. The main conclusion was that PNAR was able to discern levels of neutron multiplication; the results of the Fugen experiments were much less dynamic than is expected for typical commercial fuel setups because (1) the Fugen fuel contained little fissile material as the fuel was irradiated in heavy water, (2) the water gap around the fuel was approximately triple that which is expected for commercial fuel [9].

The PNAR implementation planned for Finland, an implementation that combines (a) a  $^3\text{He}$  detector tube and polyethylene surrounded by Cd and (b) a low multiplying section produced with a Cd-liner, lends itself to a conceptual discussion of the PNAR physics. The only significant difference in the measured count rate for a section of fuel measured in both the high and low multiplying sections, is the counts resulting from the multiplication caused by the neutrons that were absorbed in the Cd-liner. The counts produced by the neutrons not absorbed in the Cd-liner are in both the numerator and denominator of the PNAR Ratio so these high-energy neutrons that are unaffected by the Cd-liner create a PNAR Ratio of 1.0; any deviation from 1.0 is due to counts produced by chain reactions initiated by neutrons that are absorbed by the Cd-liner. Because the PNAR signal is produced by the neutrons returning into the fuel with an energy below the Cd-cutoff energy of  $\sim 0.5$  eV, the PNAR technique is sometimes described as interrogating the fuel with low energy neutrons from the location of the Cd-liner.

There are two options for implementing the Finnish conceptual PNAR design: (1) either the Cd-liner is moved in and out of position to create the high and low multiplying setups, or (2) the fuel is moved between two detector sections for which one section is high multiplying and the second section is low multiplying. The two sections in this later case could be identical in all ways except for the presence of a 0.5 mm thick sheet of Cd approximately 0.5 m long in the axial direction for the low multiplying section. This latter option can be implemented by putting the low and high multiplying sections on top of each other; in this case, the fuel is moved vertically about a meter between the two measurements to assure the same section of fuel is measured in both detector sections. The latter case has more flexibility to change the multiplication in that more than just Cd can be changed to differentiate between the two sections. In the case of Finland, moving the Cd-liner is expected. This has the benefit of requiring only one detector bank and the PNAR measurement can be completed without moving the fuel. As a result, the PNAR measurement can be completed during the same time as the PGET measurement.

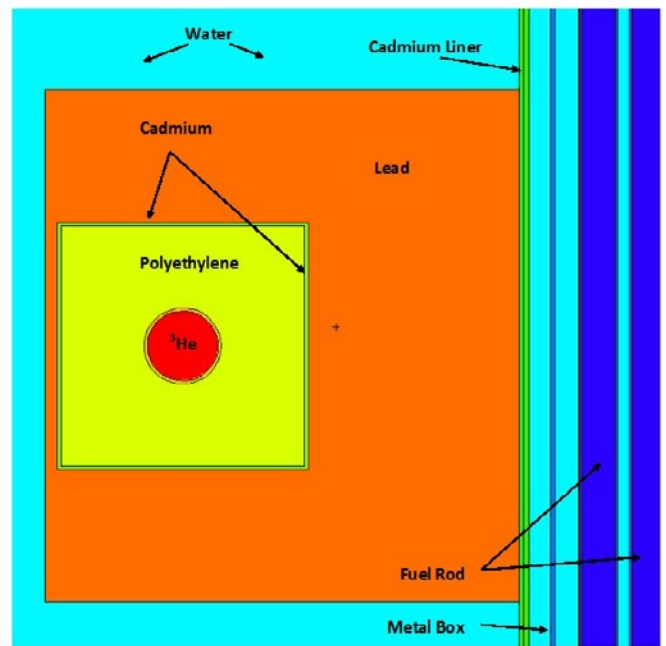
### 4. Passive neutron albedo reactivity hardware

The Radiation and Nuclear Safety Authority (STUK) of Finland commissioned the conceptual design of a PNAR instrument as part of an NDA System designed to meet the safeguards and safety needs of Finland in the context of spent fuel encapsulation and geological disposal. The geometry of the PNAR instrument needs to be adapted to the dimensions of each fuel type. In the case of Finland, because different fuel types reside at different facilities, BWR-specific and VVER-specific designs were developed.

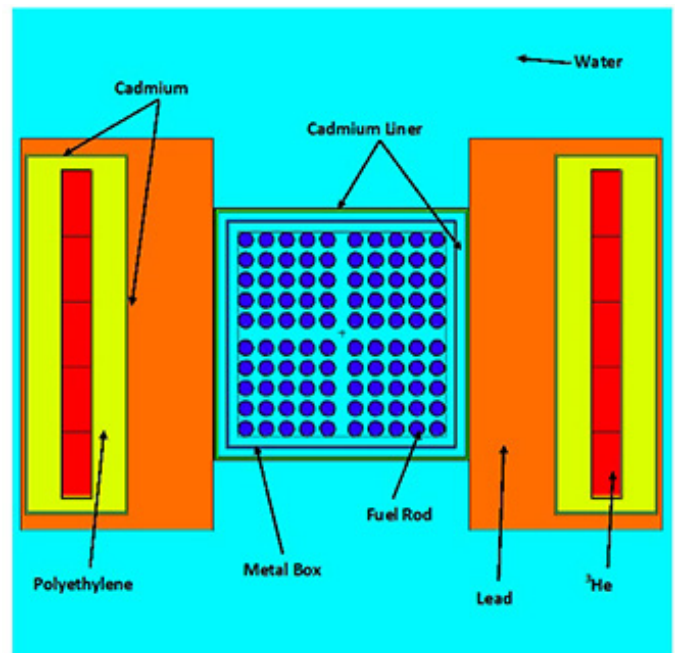
In Figure 1 and Figure 2, two cross-cutting images of the conceptual design for BWR assemblies are illustrated. The size of some of the key components of the PNAR detector are the following: The  $^3\text{He}$  tubes are 17.4 mm in diameter with a fill pressure of 6 atm and an active length of 0.2 m. The lead, needed to reduce the gamma dose to the  $^3\text{He}$  tubes, is 52 mm thick at its thickest. All Cd layers are 1 mm thick. The Cd-liner surrounds the fuel and is 0.74 m long; shorter length liners are under investigation. The starting point for the design involved calculating the amount of lead needed to keep the gamma dose to the  $^3\text{He}$  tube below 0.2 Gy/hr limit for a 0.35 m active length tube [10]. The second step involved optimizing the polyethylene near the tube for the largest count rate possible. The final step involved making sure the count rate did not exceed the recommended maximum count rate of  $5 \times 10^4$  count/s for the selected tube [11].

In this section, the hardware used to implement the PNAR concept is described with an emphasis placed on how the PNAR design partially fulfils characteristic (d) of the ASTOR Experts Group, which recommends the use of robust, low maintenance hardware. The fulfilment of this recommendation is addressed while describing the key component of the PNAR instrument by comparing, when applicable, the hardware used in implementing the PNAR concept to that of a Fork detector [12, 13]. The Fork detector was selected because it is a robust safeguards instrument, which has been used in the field for several decades:

1. With the inclusion of an ion chamber, the PNAR instrument as designed is effectively a high efficiency Fork detector with reduced positioning uncertainty due to (a) detectors located on each of the 4 sides of the fuel and (b) a smaller water gap between the fuel and the detector than with a Fork. Note that the second set of detectors, which are located above and below the assembly in Figure 2, are not visible because they are located 0.1 m below the illustrated detectors. STUK is currently investigating a PNAR design with all detectors on one axial level.
2. The key to implementing the PNAR concept is the “Cd-liner” that is depicted between the fuel and lead section of the detector. Note that this Cd-liner is only present for one of the two PNAR measurements.
3. A second layer of Cd, around the polyethylene, surrounds each  $^3\text{He}$  tube. This Cd ensures that the detector only detects epithermal and fast neutrons from the fuel; in some Fork detector designs, Cd is also used for this purpose. A more uniform spatial sensitivity across the assembly is achieved by detecting these higher energy neutrons [5].



**Figure 1.** Vertical cross-sectional view of the BWR PNAR detector along one side of a BWR fuel assembly. The lead shielding is 0.128 m vertically and 0.117 m horizontally. Proportions are accurate.



**Figure 2.** Horizontal cross-sectional view of the BWR PNAR detector relative to a 10x10 BWR fuel assembly, which is 0.126 m on a side. The detector units illustrated here span 0.128 m in the vertical direction below the vertical midplane of the detector. An identical set of detectors span the 0.128 m in the vertical direction above the midplane; these additional detectors look identical to those depicted here except they are rotated 90 degrees around the center of the fuel. Proportions are accurate.

There are three aspects of the PNAR hardware, as simulated, that deviate from the hardware of a Fork detector:

- <sup>3</sup>He tubes are used instead of fission chambers. This selection was made to obtain the desired precision for typical fuel assemblies in two minutes or less. Fission chambers or boron tubes could be used if longer count times are acceptable. The lead is present to reduce the gamma dose to the <sup>3</sup>He tubes; the lead would not be necessary if fission chambers are used; yet given that the installation is permanent, the weight of lead is not a significant concern.
- The instrument is designed with the expectation that the fuel will be inserted from above into the detector. The detectors are located on the 4 sides of the assembly; it is expected that the instrument will not measure locations near the ends of the assembly. This selection was made to reduce the sensitivity to anisotropy in the assembly burnup. If such an uncertainty is not too large, a reduction to two detectors on opposite sides of the fuel is acceptable.
- The presence of a 1 mm thick, 0.74 m long axial Cd-liner. This sheet of metal is the sole PNAR component that is not necessary for a Fork detector.

## 5. Simulated passive neutron albedo reactivity signature

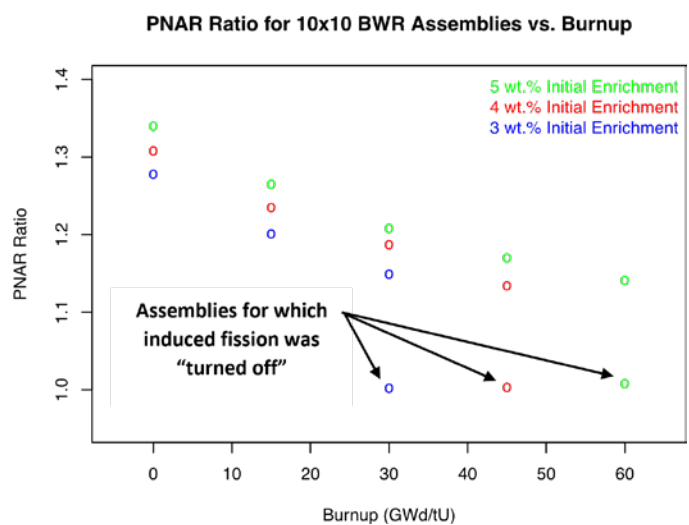
To access the capability of the PNAR detector to measure spent fuel, the PNAR ratio was calculated using the average isotopic content of 12 assemblies that span a range of initial enrichment (3, 4 and 5 wt.%), burnup (15, 30, 40 and 60 GWd/tU) and cooling time (20, 40 and 80 years) values. The Monte Carlo N-Particle Code (MCNP6™), Version 6 [14] was used for the PNAR simulations while the isotopic mixture of the various assemblies was produced by the MonteBurns code [15] as part of the Next Generation Safeguards Initiative [16, 17].

Figure 3 shows the calculated PNAR ratio versus burnup for 12 different assemblies in fresh water. Two simulations were run to calculate each data point, once with the Cd-liner in place and once without the Cd-liner. All data points in Fig. 3 are for fuel with a 20-years cooling time.

The PNAR Ratio values for all the data points with ratios above 1.1 were simulated in the standard manner, meaning that all neutrons, and subsequent reactions that they may cause, were followed until the neutrons were either absorbed or left the extremities of the simulation; any nuclear reactions that produced additional neutrons, such as induced fission, were followed through to fruition.

For the three assemblies considered to be nearly fully irradiated, given their initial enrichment and burnup values, which are the assemblies with PNAR Ratios of about 1.14, additional simulations were performed to calculate the PNAR ratio for the case when no induced fission could

take place. The three assemblies are the following: (a) 3 wt.%, 30 GWd/tU, (b) 4 wt.%, 45 GWd/tU, (a) 5 wt.%, 60 GWd/tU; these assemblies are labelled separately in Fig. 4. For these 3 assemblies, induced fission reactions became absorption reactions. This is a useful exercise because it indicates the signal expected if all the fuel were replaced with a non-multiplying material. This change in the simulation was accomplished by adding the "NONU" card to the simulation. The calculated PNAR Ratio for each of these assemblies with the NONU card is 1.002, 1.003 and 1.008, respectively. The absolute value of the uncertainty on the PNAR ratio, propagated from the MCNP6™ statistical uncertainty, is 0.003 for all points. The vertical extent of each data point in Figure 3 is approximately 4 times the propagated statistical uncertainty calculated with MCNP6™.



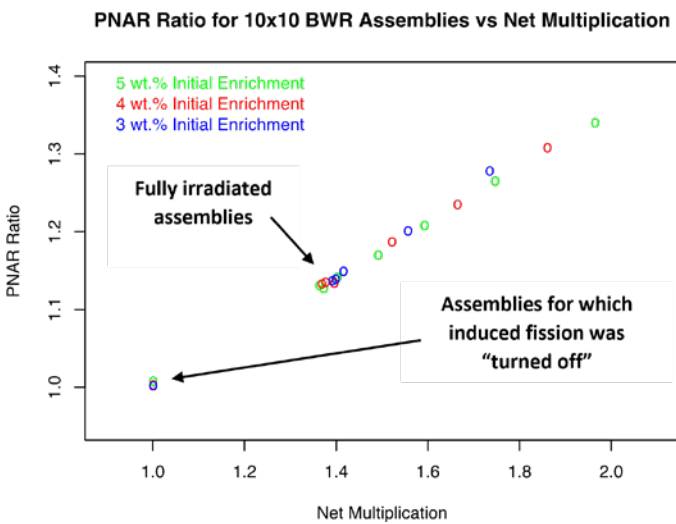
**Figure 3.** the BWR PNAR Ratio, simulated with fresh water, is illustrated as a function of burnup for 12 assemblies of various initial enrichments and burnup values; 3 assemblies were simulated using the nonu card. The cooling time is 20 years. The vertical extent of each data point is approximately equal to 4-sigma of statistical uncertainty arising from the simulated statistics.

The following are key points concluded from Figure 3:

- The change in the PNAR Ratio with increasing burnup is a smooth decreasing function of burnup for a given initial enrichment. If an assembly starts with more potential nuclear energy, it will be measured to have an elevated PNAR ratio when fresh.
- Fully irradiated assemblies, regardless of their initial enrichment, are expected to have nearly the same PNAR Ratio.
- The change in the PNAR Ratio between a fresh 4 wt.% assembly and a 4 wt.% assembly that was irradiated to 45 GWd/tU is approximately the same as the change in the PNAR Ratio calculated between a fully irradiated assembly and an assembly for which all the pins were replaced with non-multiplying material.



In Figure 4, the PNAR Ratio is graphed as a function of the “net multiplication;” which is calculated by the MCNP6 code by taking the ratio of the number of neutrons started in the fuel to the number of neutrons followed during the course of the simulation. Note that the net multiplication is calculated for the case of neutrons starting from all the pins in the assembly with the energy sampled from a Watt fission spectrum. The data points in Figure 4 include all the same data points illustrated in Figure 3 as well as 6 additional assemblies. These additional assemblies are for the three nearly fully irradiated assemblies already discussed; however, the isotopic content was “aged” to represent that which is expected for cooling times of 40 and 80-years, in addition to the 20-year cooling time case from Figure 3. The main point for including these assemblies is to show that the 9 fully irradiated assemblies occupy a small area of the overall parameter space; additionally, among these 9 assemblies, the 3 assemblies with the same initial enrichment but cooling times of 20, 40 and 80 years are clustered in an even smaller area.



**Figure 4.** the PNAR Ratio is graphed as a function of the net multiplication. 3 assemblies with cooling times of 40 years and 3 assemblies with cooling times of 80 years were included compared to Figure 3. All these longer-cooled assemblies group around a net multiplication value of 1.41 and a PNAR Ratio of 1.14 and are labeled as “fully irradiated assemblies.”

The conclusions drawn from Figure 4 are the following:

1. Regardless of initial enrichment, burnup or cooling time, there is a smooth relationship between the PNAR Ratio and net multiplication.
2. There is a change of between 0.13 and 0.14 in the PNAR Ratio between any irradiated assembly and a non-multiplying assembly.
3. Almost all assemblies to be measured at an encapsulation facility will be fully irradiated. Hence, a near constant PNAR Ratio will be measured for all these assemblies; if the current simulations are representative, that value will be around 1.14.

4. From Figure 4, we can see that the impact of cooling time is relatively small, the 9 fully irradiated assemblies all have net multiplications values of around 1.4 and PNAR Ratios of around 1.14.

Because the uncertainty of the PNAR instrument is connected to how useful the instrument can be, the results from a separate report examining the anticipated uncertainty are summarized here [18]. In that report, the uncertainties due to the following were examined: (a) assembly position in the instrument, (b) counting statistics given a total measurement time of 5-minutes, (c) estimated uncertainty given non-uniform irradiation. The end conclusion was that a rough estimate of the 1-sigma uncertainty in the PNAR Ratio is anticipated to be +/- 0.005 for a typical BWR assembly (32 GWd/tU, 40-year cooled) given a 5 minute total count time; while the coolest assemblies (17 GWd/tU, 60-year cooled) will likely need a 20-minute count time to obtain a similar uncertainty. To put the 0.005 value in some context, a 0.005 variation in the PNAR Ratio corresponds to the change in the multiplication caused by a burnup variation of 1.4 GWd/tU. Additionally, if a fully irradiated assembly were replaced with a non-multiplying assembly, then the PNAR Ratio should change from 1.142 to 1.002 for a net change of 0.140; this represents a change of 28 sigma; the main point being that a significant removal of fissile material from the assembly will be easily detected.

A point worth emphasizing is that the PNAR technique fulfills the recommendations of the IAEA ASTOR group that the NDA system be “capable of measuring assembly neutron multiplication.”

## 6. Merit of an integrated NDA system

The ASTOR Experts Group recommendations (b) and (e) are discussed together within the context of the merits of an integrated NDA system. Characteristic (b) involves verifying the declaration with the measured signatures, while characteristic (e) involves creating an NDA system that is difficult to trick.

The integrated Finnish NDA system suggested by STUK has the following measured signatures:

1. Relative distribution of the gamma ray emission within a horizontal cross-section of an assembly with pin level resolution using the PGET instrument. The PGET instrument [2], after recent refurbishment by IAEA [3], has shown a capability for automated detection of single or multiple missing pins in BWR and VVER-440 fuel. However, improved image reconstruction and analysis techniques are still required [4].
2. Absolute gross gamma intensity as measured by ion chambers, which are built into the PNAR instrument, and the absolute <sup>137</sup>Cs count rate as measured by the CZT detectors of the PGET instrument.

3. Absolute neutron count rate as measured with the PNAR detector and the boron-tube neutron detectors in the PGET instrument.
4. Absolute neutron multiplication as measured with the PNAR detector.

The analytic approach of the Finnish NDA system has two separate parts: (a) One part analyses the PGET information to create relative-intensity gamma-ray images. (b) The second analytic approach uses the information declared by the state to calculate the multiplication, absolute gamma and neutron source terms. Considerable research along these lines was performed by Euratom and Oak Ridge National Laboratory researchers [19]. Their analytic approach uses both SCALE [20] and MCNP6™ to calculate the assembly multiplication as well as the neutron and gamma flux from the fuel to any relevant detectors.

Focusing on the SCALE/MCNP6™ portion of the analysis for the Finnish case, the gamma intensity measured by both the CZT and ion chambers can be compared to the values calculated by SCALE and MCNP6™ from the declaration. Similarly, the total neutron count rate and the neutron multiplication can be calculated from the declaration and compared to the measured values. In the case of the measured multiplication, either the PNAR Ratio itself could be calculated or a calibrated correlation between the net multiplication and the PNAR Ratio, as illustrated in Figure 4, could be used.

An additional possible part of the analysis could involve the calculation of the neutron source term. The neutron source term is equal to the total neutron emission divided by the net multiplication. Given the relationship between the net multiplication and the PNAR Ratio illustrated in Figure 4, the intensity of the neutron source term can be calculated.

In summary, ASTOR characteristic (b), which involves verifying the State declaration, will be satisfied for the long-cooled fuel of interest in Finland by measuring the following characteristics of the fuel:

1. Both the absolute gross gamma intensity and  $^{137}\text{Cs}$  count rate will be measured and compared to simulation; both signatures vary with a 30.2 year half-life for longer cooling times.
2. Total neutron count rate will be measured with the PNAR detector and with the boron-tube neutron detectors in the PGET instrument and compared to simulation. This signature primarily varies with the 18.1 year half-life of  $^{244}\text{Cm}$ .
3. Neutron multiplication will be measured with the PNAR detector and compared to simulation; a signature that is expected to vary by ~5% as the fuel ages from 20 to 80 years [21].

Combining the above list with a total weight measurement and the 2-dimensional, pin-localizing, image of the pin gamma ray intensities produced by PGET, the challenge a 'would

be' proliferator has in tricking the integrated NDA system is imposing. This proliferator would need to do all the following:

1. Emit gamma rays with the correct energy/energies and relative intensity from all the pins in a BWR or a VVER-440 assembly.
2. Emit  $^{137}\text{Cs}$  photons and/or create an absolute current in the ion chambers that is consistent with the initial enrichment, burnup and cooling time of the declaration.
3. Produce two specific and related neutron count rates when the assembly is measured in two different neutrons reflecting setups. The relative intensity of the count rates, which is the indication of the level of multiplication, must be consistent with the initial enrichment, burnup and cooling time of the declaration.
4. Keep the assembly weight within the uncertainty limits of the weight measurement.

Given (a) the time varying complexity of the signatures ( $T_{1/2} = 30.2$  years for 662 keV photons from  $^{137}\text{Cs}$ ,  $T_{1/2} = 18.1$  years for total neutrons given the dominance of  $^{244}\text{Cm}$ , while the multiplication remains nearly constant as function of time) and (b) the pin level resolved image from PGET, the proposed NDA system is "difficult to trick with pin substitution"; hence ASTOR recommendation (e), is satisfied.

## 7. Conclusion

The PNAR instrument is a robust instrument made from mature off-the-shelf hardware. Combined with a PGET, the integrated instrument satisfies all the characteristics suggested by the NDA Focus Group convened by the IAEA as part of the ASTOR Experts Group: (a) For pin level detection, PGET is expected to be able to detect single and multiple missing pins in BWR and VVER-440 fuel. If there will be cases where this detection capability is not fully assured, the integrated NDA system will detect the absence of fuel when significant inconsistencies are detected among the multiplication, total neutron or gamma signatures. (b) For declaration verification, the measured multiplication, neutron and gamma signatures will be compared to the calculated values for each of these signatures that used the declaration as input values. (c) The multiplication will be measured by a PNAR instrument calibrated with known assemblies. (d) The hardware is expected to be robust and low-maintenance. (e) Given the range of measured signatures: spatial gamma ray emission, total neutron and gamma count rates, multiplication and assembly weight, combined with declaration-based analysis; the overall system is difficult to trick with pin substitution.

## 8. References

- [1] ASTOR Group Report 2011-2016, "Technologies Potentially Useful for Safeguarding Geological Repositories," International Atomic Energy Agency, STR-384, Vienna, Austria 2017.

- [2] T. Honkamaa, F. Levai, A. Turunen, R. Berndt, S. Vaccaro and P. Schwalbach, "A Prototype for Passive Gamma Emission Tomography," Symposium on International Safeguards: Linking Strategy, Implementation and People, Vienna, Austria, 2014.
- [3] M. Mayorov, T. White, A. Lebrun, J. Brutscher, J. Keubler, A. Birnbaum, V. Ivanov, T. Honkamaa, P. Peura and J. Dahlberg, "Gamma Emission Tomography for the Inspection of Spent Nuclear Fuel," to be published in the IEEE NSS Conference Record, 2017.
- [4] T. White, M. Mayorov, N. Deshmukh, E. Miller, L.E. Smith, J. Dahlberg and T. Honkamaa, "SPECT Reconstruction and Analysis for the Inspection of Spent Nuclear Fuel," to be published in the IEEE NSS Conference Record, 2017.
- [5] D.M. Lee and L.O. Lindqvist, "Self-Interrogation of Spent Fuel," Los Alamos, LA-9494-MS, UC-15, 1982.
- [6] J.L. Conlin, S.J. Tobin, J. Hu, T.H. Lee, H.O. Menlove, "Passive Neutron Albedo Reactivity with Fission Chambers," LA-UR-11-00521, 2010.
- [7] A. Bolind, "Development of an Analytical Theory to Describe the PNAR and CIPN Nondestructive Assay Techniques," Annals of Nuclear Energy, vol. 66, pp. 167-176, 2014.
- [8] Menlove, H.O., Beddingfield, D.H., Passive Neutron Reactivity Measurement Technique, Los Alamos National Laboratory document LA-UR-97-2651 (May 1997).
- [9] Eigenbrodt, Julia M (2016). **Spent Fuel Measurements: Passive Neutron Albedo Reactivity (PNAR) and Photon Signatures. Doctoral dissertation, Texas A & M University.** Available electronically from <http://hdl.handle.net/1969.1/156845>.
- [10] D.H. Beddingfield, N.H. Johnson and H.O. Menlove, "<sup>3</sup>He Neutron Proportional Counter Performance in High Gamma-Ray Dose Environments," *Nuclear Instruments & Methods in Physics Research, Section A*, vol. 455, pp. 670-682, 2000.
- [11] Personal communication with Nathan Johnson, GE Reuter Stokes, Twinsburg, OH, USA, 2017.
- [12] P.M. Rinard and G.E. Bosler, "Safeguarding LWR Spent Fuel with the Fork Detector," LA-11096-MS, Los Alamos National Laboratory, New Mexico, USA, 1988.
- [13] A. Tiita et al., "Investigation of the Possibility to Use Fork Detector for Partial Defect Verification of Spent LWR Fuel Assemblies," ISSN 0785-9325, ISBN 951-712-590-9, Helsinki, 2002.
- [14] T. Goorley et al., "Initial MCNP6 Release Overview," Nuclear Technology, vol. 180, pp. 298-315, Dec. 2012.
- [15] H.R. Trellue, M.L. Fensin, J.R. Richard, and J.L. Conlin, "Description of the Spent Nuclear Fuel Used in the Next Generation Safeguards Initiative to Determine Plutonium Mass in Spent Fuel," Los Alamos National Laboratory, LA-UR 11-00300, 2011.
- [16] M.A. Humphrey, S.J. Tobin and K. Veal, "The Next Generation Safeguards Initiative's Spent Fuel Nondestructive Assay Project," Journal of Nuclear Material Management, vol. 3:XL, 2012.
- [17] S.J. Tobin et al., "Experimental and Analytical Plans for the Non-Destructive Assay System of the Swedish Encapsulation and Repository Facilities," Symposium on International Safeguards: Linking Strategy, Implementation and People, Vols. IAEA-CN-220-238, 2014.
- [18] S.J. Tobin, P. Peura, T. Honkamaa, P. Dendooven, M. Moring, and C. Bélanger-Champagne, "Passive Neutron Albedo Reactivity in the Finnish Encapsulation Context," Finnish Radiation and Nuclear Safety Authority, ISBN 978-952-309-406-2, Helsinki Finland, Dec. 2017.
- [19] S. Vaccaro, J. Hu, J. Svedkauskaite, A. Smejkal, P. Schwalbach, P. De-Baere and I.C. Gauld, "A New Approach to Fork Measurements Data Analysis by RADAR-CRISP and ORIGEN Integration," IEEE Transactions on Nuclear Science, vol. 61, no. 4, pp. 2161-2168, March 2014.
- [20] SCALE: "A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design," Oak Ridge National Laboratory, Oak Ridge, TN, ORNL/TM-2005/39, June 2011.
- [21] S.J. Tobin, M.L. Fugate, H.R. Trellue, T. Burr, P. De-Baere, P. Jansson, P. Schwalbach, A. Sjolund and S. Vaccaro, "Research into Measured and Simulated Nondestructive Assay Data to Address the Spent Fuel Assay Needs of Nuclear Repositories," in Institute of Nuclear Material Management, Atlanta, GA, 2016



# Ultrasonic Investigation of the Welding Area of Copper Canisters for Spent Nuclear Fuel

C. Clementi<sup>1,2</sup>, F. Littmann<sup>1</sup>, L. Capineri<sup>2</sup>

<sup>1</sup> Nuclear Security Unit, Joint Research Centre, European Commission, Ispra (VA), Italy

<sup>2</sup> Dept. Information Engineering, University of Florence, Florence, Italy

E-mail: chiara.clementi@ec.europa.eu, francois.littmann@ec.europa.eu, lorenzo.capineri@unifi.it

## Abstract:

Geological repositories will be built in several countries for the long term storage of spent nuclear fuel. Sweden and Finland, for example, foresee the encapsulation of the spent fuel assemblies in copper canisters with iron inserts to be deposited in tunnels excavated about 500 metres underground in the bedrock. During the transport of canisters and the storage in the final repository, the Continuity of Knowledge (CoK) of spent fuel must be kept. Safeguards requirements suggest the implementation of a unique identification tag on copper canisters, strongly reliable and secure against falsification attempts. This paper presents a solution for the canisters' authentication issue, using an innovative ultrasonic investigation of the internal gap between copper lid and canister tube. It provides a fingerprint strictly related to material features after the friction stir welding process. This fingerprint can be combined with the ultrasonic response of a unique code made by chamfers machined on the copper lid inner surface. The angular matching between the two fingerprints (internal gap response and chamfers' code

reading) realizes a third signature more robust and reliable. A potential practical implementation of this solution is described experimentally in this paper.

**Keywords:** copper canisters; authentication; ultrasounds; encapsulation plant; geological repository.

## 1. Introduction

The spent fuel coming from nuclear reactors is an environmental hazard and it must be safely managed according to Safety, Security and Safeguards requirements. Several countries, among which Sweden and Finland, are planning to construct geological repositories for the storage of their spent fuel. According to this new approach, the fuel will be encapsulated in copper canisters with iron inserts, deposited in tunnels 500 m underground and covered by bentonite clay. Geological repositories, in fact, would be able to insulate the spent fuel from human beings and the environment without requiring supervision or maintenance after closing [1]. Figure 1 shows the structure of the geological repository planned to be built in Sweden.

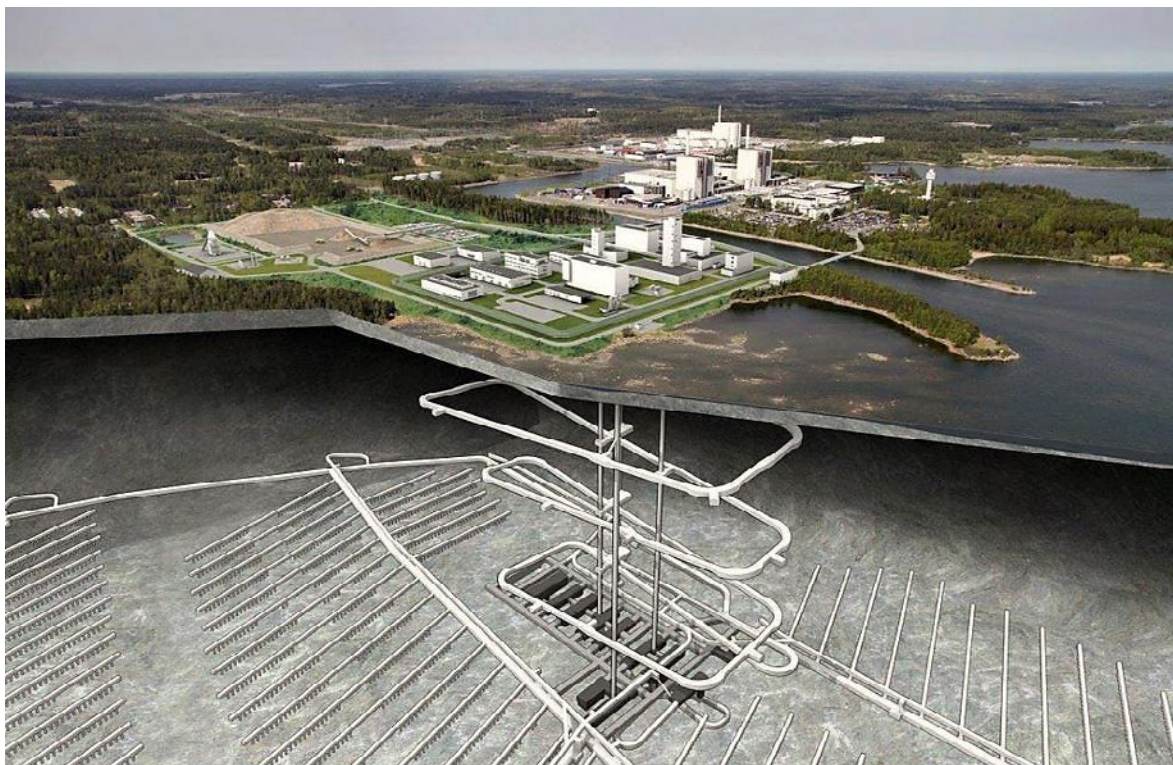


Figure 1. Design of the geological repository proposed for Sweden with its 66-kilometre of tunnels. Source: SKB.



The fuel stored in underwater storage ponds is moved to the encapsulation plant where it is introduced in copper canisters. Canisters are then sealed by Friction Stir Welding (FSW) then, for Sweden, they are inserted into transport casks sent by ship to the final repository. For Finland, the encapsulation plant will be built just above the geological repository, so no transport casks needed. The analysis of threat, diversion strategies and safeguards requirements for a geological repository stressed the importance of keeping the Continuity of Knowledge (CoK) of spent fuel during transport from the encapsulation plant to the final storage or during the storage of canisters before the deposition in tunnels. Containment and Surveillance (C/S) measures have to be implemented in the encapsulation process and on the transportation cask [2]. Moreover, the International Atomic Energy Agency (IAEA) and the European Atomic Energy Community (EURATOM) recommended to use canister identification to support the CoK of spent fuel [3].

Engraving of the canister's external surface is a practical and simple solution but may affect the long-term integrity of the container and then must be carefully considered. Moreover an engraving on the external surface can be easily reproduced and then it is not secure against falsification attempts. Considering alternative solutions, the most common technologies used for tagging of nuclear items include seals, Radio Frequency Identification (RFID) tags, SERS (Surface-Enhanced Raman Scattering) -Active Nanoparticle Aggregates tags, Tungsten-based identifier, reflective laser scanning tags, reflective particle tags (RPT) and ultrasonic systems [4]. All these techniques could potentially provide a unique identification for canisters but only ultrasounds could satisfy both the requirements of identification and authentication at the same time [5]. In 2015 the SSM asked the Joint Research Centre (JRC) of the European Commission for a feasibility study on copper canisters identification by ultrasounds. This research has been carried out by the Seals and Identification Laboratory (SILab) of the JRC in Ispra (Italy) in collaboration with the Department of Information Engineering at the University of Florence (Italy). The SILab has a long time experience on ultrasonic techniques applied on bolt seals with artificial cavities made of stainless steel washers with cavities, giving a fingerprint from the reflection of unique patterns [6]. Concerning copper canisters, the geometry and dimension of the container are much bigger than a seal; therefore, an adaption of the ultrasonic methods should be implemented.

The following chapters describe the development of the new solution for the identification and authentication of copper canisters by ultrasounds. The first part illustrates the basic idea of the new method and the preliminary studies carried out on copper specimens to verify the propagation of ultrasounds in the material. Afterwards the experimental tests carried out at the SKB's Canister laboratory in Oskarshamn (Sweden) are reported and results are analysed.

## 2. Ultrasonic authentication of copper canisters

The method used for labelling copper canisters must be tamper indicating by a unique identity (identification) and must provide evidence of counterfeiting or duplication (authentication). For this purpose, the method developed by the SILab is based on the combination of two fingerprints, one artificial and one natural. The first is realized by machining chamfers on the inner surface of the copper lid, creating a unique code readable from the outside by an inclined ultrasonic transducer with water poured on the lid. The study of this identification method has been already described by the authors in [7] and [7]. Results of simulation and experimental tests made on a reduced scaled copper lid with chamfers, pointed out the possibility to acquire ultrasonic echoes with a good signal to noise ratio. The code made by chamfers is then clearly detectable by ultrasounds but it is not robust enough in case of falsification or duplication attempts. Therefore, an authentication signature is necessary to verify the originality of canisters. This paper is focused on the study of an authentication method based on the ultrasonic investigation of the natural fingerprint, intrinsically contained in copper canisters after the FSW process. The general idea of the method and its validation is described in the following paragraphs.

### 2.1 The method

The nuclear spent fuel to be stored in geological repositories is preserved by an insert of nodular cast iron that gives mechanical stability and an outer copper shell which provides protection against corrosion. The copper canister is about 5 metres in length and 1 metre in diameter [9]. The minimum copper thickness is 50 mm to fulfil mechanical design requirements [10] (Figure 2).

After the encapsulation of the fuel, a lid is welded onto the canister by FSW. A rotating tool is plunged between the

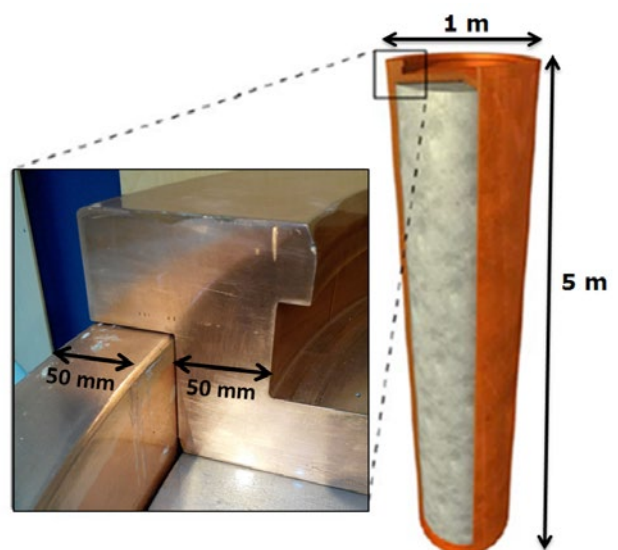
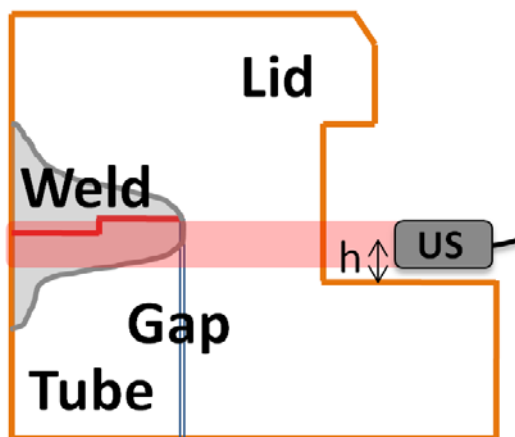


Figure 2. Copper canisters dimensions.

pieces to be welded and makes the material plastic by the heat generated with the friction. During the tool movement the two surfaces are stirred together and a joint is realized [11]. The weld joint design includes a gap between the lid and the canister. This gap represents a discontinuity in the material, detectable by an ultrasonic transducer placed beneath the welding line (Figure 3).

The amplitude response acquired by the ultrasonic testing around the circumference of the welded canister could be used as a fingerprint, unique and different to each canister.

Before the evaluation of the amplitude response of the welding area of a full scale copper canister, preliminary studies were carried out on copper flanges, i.e. slices of the copper lid already welded onto the tube. The first tests implemented by the manual displacement of a contact transducer (V111-RM Fingertip Contact Transducers - Panametrics) revealed a series of fingerprints such as those illustrated in Figure 4. The probe is moved along arcs 380 mm wide positioned at different heights ( $h=25, 23, 21, 19, 17, 15$  mm) from the bottom surface of the lid (see Figure 3).

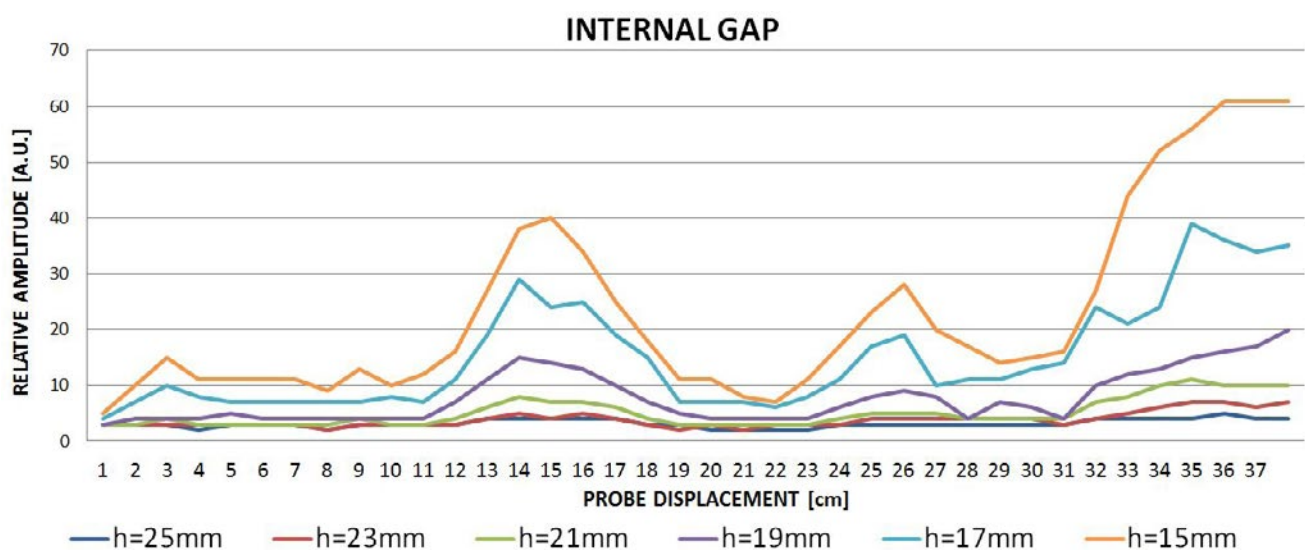


**Figure 3.** Position of the ultrasonic transducer for the investigation of the gap between copper lid and tube.

The variation of the ultrasonic amplitude echo is more marked at  $h=15$  mm where the gap is clearly detectable. Moving the probe at higher positions the amplitude of the received echo is lower. This means that the gap is disappearing from the focus of the probe and the interface between lid and tube is perfectly welded. As a consequence, the presence of a discontinuity in the material detectable by ultrasounds is demonstrated. However, it is important to remember that flanges are just laboratory samples and that the variation of patterns in a real canister coming from a production-like process could be lower. Therefore experimental tests on full scale lids already welded onto tubes must be implemented to verify the accuracy and repeatability of the method. For this purpose, the manual scanning of the sample is replaced by a reader prototype hosting a motor for an automated scanning. The description of this new system called *IDA reader* is reported in the following paragraph.

## 2.2 The IDA Reader

The IDA (IDentification and Authentication) reader is a device for the identification and authentication of copper canisters. The first prototype realized is only dedicated to the acquisition of an authentication fingerprint. The system is composed by a reader, a control box and a computer (Figure 5). The reader consists of three supporting arms made on steel, centred on the lid, and a motor devoted to the rotation of a rod holding the ultrasonic transducer (Olympus V311 with 10 MHz central frequency and 8.4 inches spherical focused). The probe is kept perpendicular to the lid surface and its distance can be adjusted from 50 mm to 25 mm, while height ( $h$ ) from 35 mm to 14 mm from the bottom of the lid. The box hosts an electronic board connected to an US-Key module (Lecoer Electronique) controlling the motor's rotation and the transmission/reception of the ultrasonic signal. Lastly the



**Figure 4.** Relative amplitude responses of the internal gap between copper lid and tube.

computer displays the software interface for the setting of testing parameters and the signal processing.

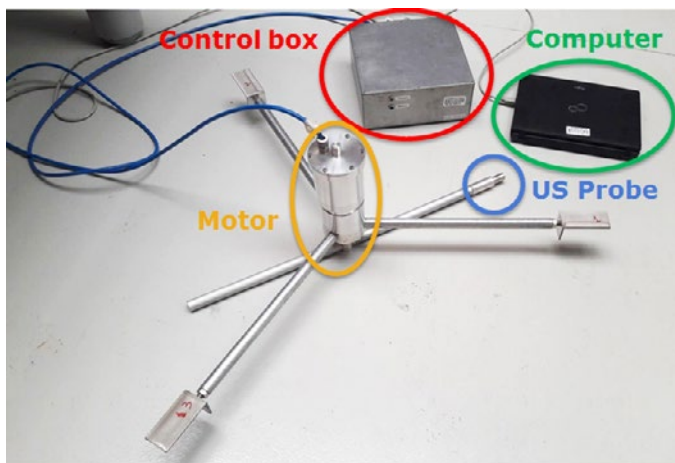


Figure 5. IDA reader prototype.

The software interface is composed by four different tabs. The first tab allows the setting of the main measurements parameters and the three gates (time windows) of interest, where echoes are expected. The second and third tabs are then focused on the acquisition of the internal gap fingerprints. The last part is centralized on the signal processing. In particular, the correlation index between two fingerprints selected from the database can be calculated and the two curves can be displayed shifted and overlapped on a chart.

The first test of the reader is carried out at the SILab facility on copper flanges arranged around a circumference reproducing the same geometry of a real copper lid welded on a tube. The three arms of the reader should be perfectly parallel to the bottom of the lid to keep the probe perpendicular to copper interface during revolution. The scan time is roughly 4 minutes, corresponding to a 360° rotation of the probe around the circumference. Before starting with tests, the temperature of water is measured through an infrared digital thermometer and the probe distance and height are set by a calliper. Depending on temperature and probe position, different time windows shall be settled in the acquisition software to ensure a good reception of the ultrasonic echoes. The set-up of measurements is shown in Figure 6.

Several tests were carried out changing the angular position of the reader's arms above flanges and the analysis of acquired echoes revealed a very high correlation between fingerprints, whether acquired without changing the set-up or adjusting the reader positions above flanges.

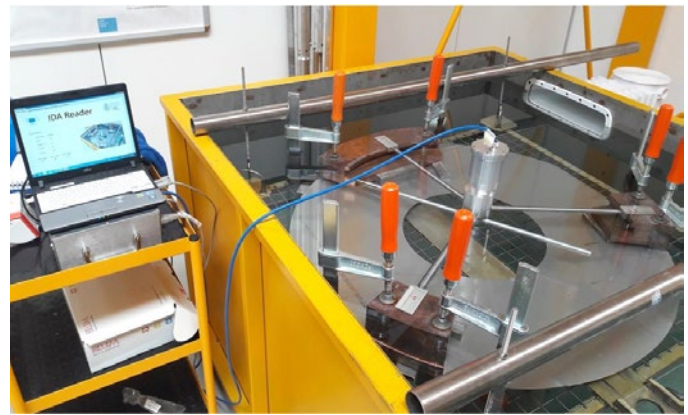


Figure 6. Set-up of measurements.

### 3. The experimental testing

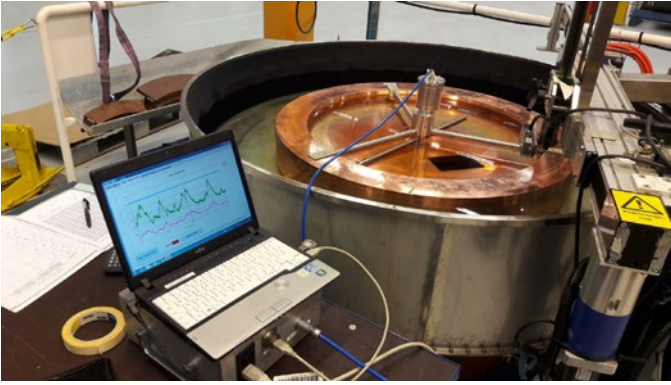
The experimental testing of the ultrasonic method for the authentication of copper canisters was done by the ultrasonic investigation of two copper lids already welded onto tubes at the SKB's Canister Laboratory. The aim of these inspections was to verify the detectability of a fluctuating fingerprint due to variations in the material after welding. Moreover, the repeatability of the method should be confirmed by the high correlation indexes between multiple acquisitions with the IDA reader. This chapter describes all the measurements implemented and their results.

#### 3.1 The ultrasonic testing at the SKB's Canister laboratory

The SKB's Canister Laboratory, situated in Oskarshamn's port, is the centre for the development of technologies that will be used for the encapsulation of spent fuel in copper canisters. The laboratory houses the SKB's prototype of the welding machine, manufactured by ESAB and used for the FWS of the lids to the canisters [12].

The ultrasonic inspections with IDA Reader were carried out on two samples of copper lids already welded onto tubes (FSWL 121 and FSWL 122). The geometries of the two original lids were slightly different therefore the uniqueness of fingerprints could not be checked. However comparisons between received echoes are useful to evaluate the efficacy of the ultrasonic method. Copper samples are placed in a tank with water (Figure 7) and four labels are arranged at 90° from each other on the upper surface of the lid to create a reference pattern for measures.





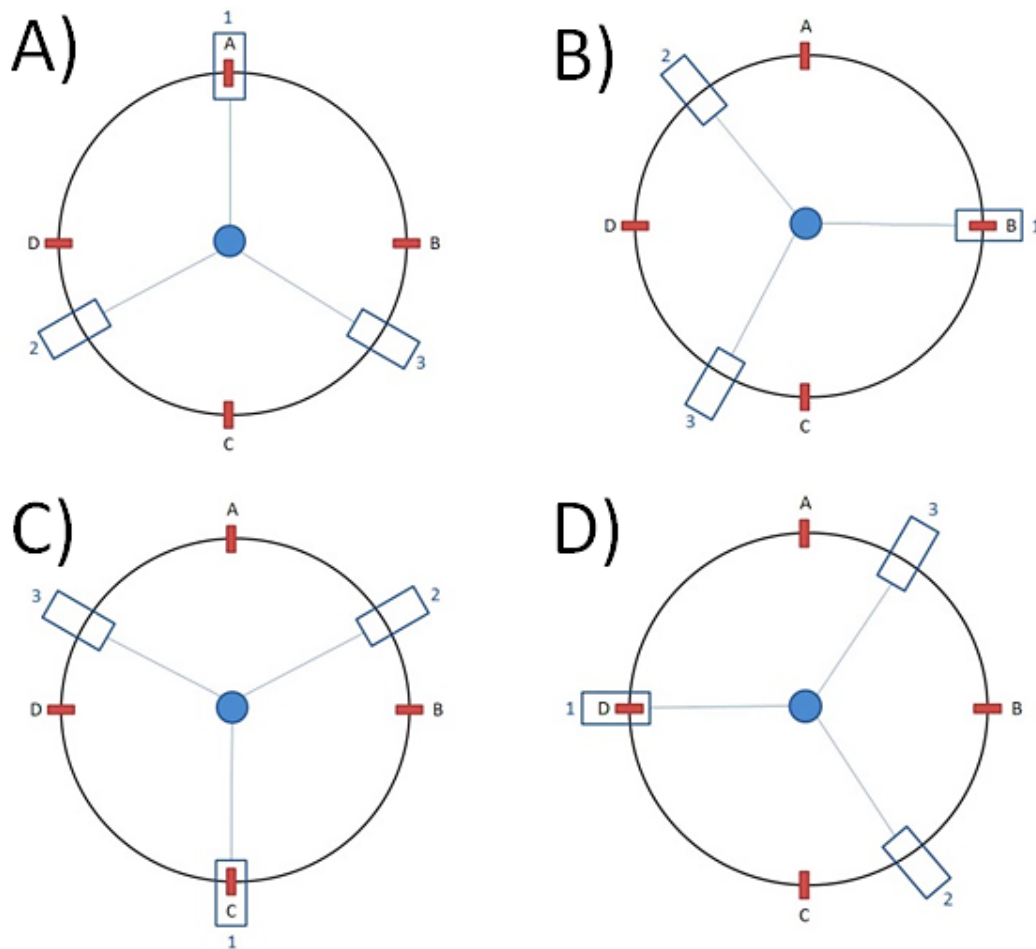
**Figure 7.** Experimental testing of the IDA reader at the SKB's Canister Laboratory.

In this way, different acquisitions are carried out changing the arms positions around the circumference. The series of measurements implemented is shown in Figure 8. For each position A, B, C, D, the height of the transducer is changed from 15 mm to 21 mm with a step of 2 mm. Through the variation of the reader position, it is possible to verify the reader ability to perform repeatable measurements. The adjustment of probe heights, instead, aims to

detect which is the variation of the ultrasonic amplitude response in the entire welding area.

### 3.2 Analysis of results

The analysis of the data acquired during the ultrasonic investigations is presented in this paragraph. The most meaningful fingerprints are reported and compared to identify the main features. The first result illustrated concerns the investigation of the FSWL 122 at 15 mm of probe height. The temperature of water during the test was 18°C, the probe distance 50 mm and the gain of the electronic receiver was 37 dB. The chart below (Figure 9) shows the ultrasonic amplitude response related to two different gates (gate 2: 82-97 $\mu$ s and gate 3: 102-117  $\mu$ s), on 360° of the lid circumference. The signal in blue is the amplitude response due to the reflection on the internal gap. In purple, instead, it is shown the double reflection of the previous signal that is temporally located at the same time windows than the external wall echo (because thicknesses of tube and lid in this region are both 50 mm). The vertical line in red marks the starting point of the weld (the entry point of the welding tool).



**Figure 8.** Sequence of measurements implemented on copper samples.



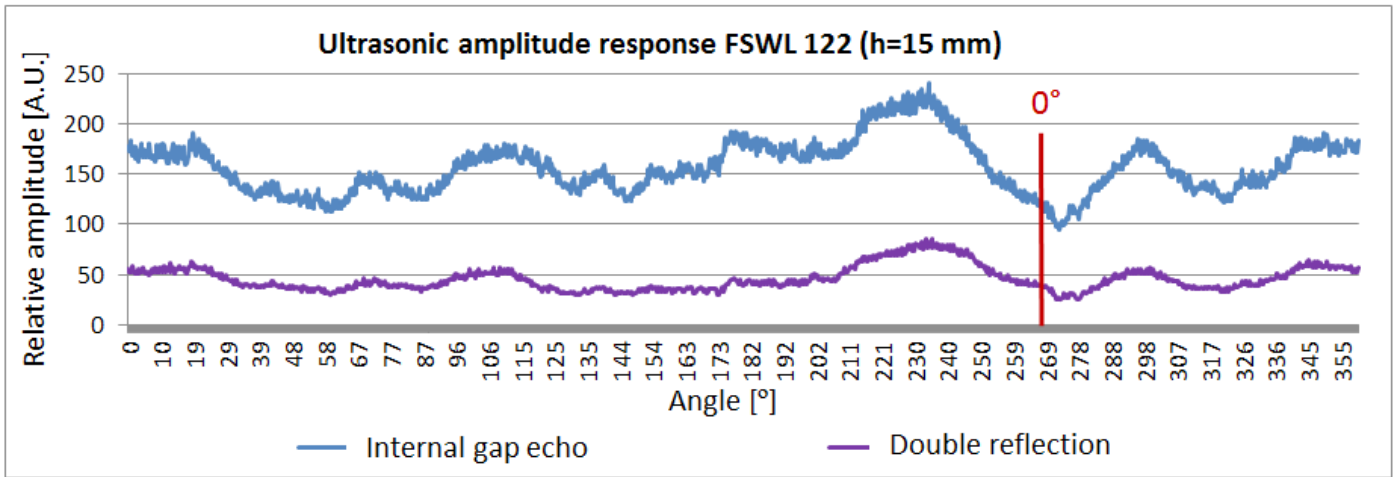


Figure 9. Ultrasonic amplitude response of FSWL 122 at 15 mm height.

The variation of the signal could be related to the different grain size in the material that generates a different attenuation pattern around the lid circumference. Several measurements are implemented changing the position of the reader above the sample and the correlation between different acquisitions is high. Therefore the repeatability of signatures is verified and signals are stable if we consider a constant water temperature, even though influence of water temperature can be easily compensated.

In order to verify that the echo in purple was exactly the repetition of the first in blue, an inspection simulation with CIVA software is implemented (Figure 10). As a result, the simulated A-signal exhibit three main echoes: the first related to the interface water/copper, the second due to the reflection on the internal gap (corresponding to the blue trace) and then a second reflection whose features (amplitude and time of flight) agrees with our results (purple trace).

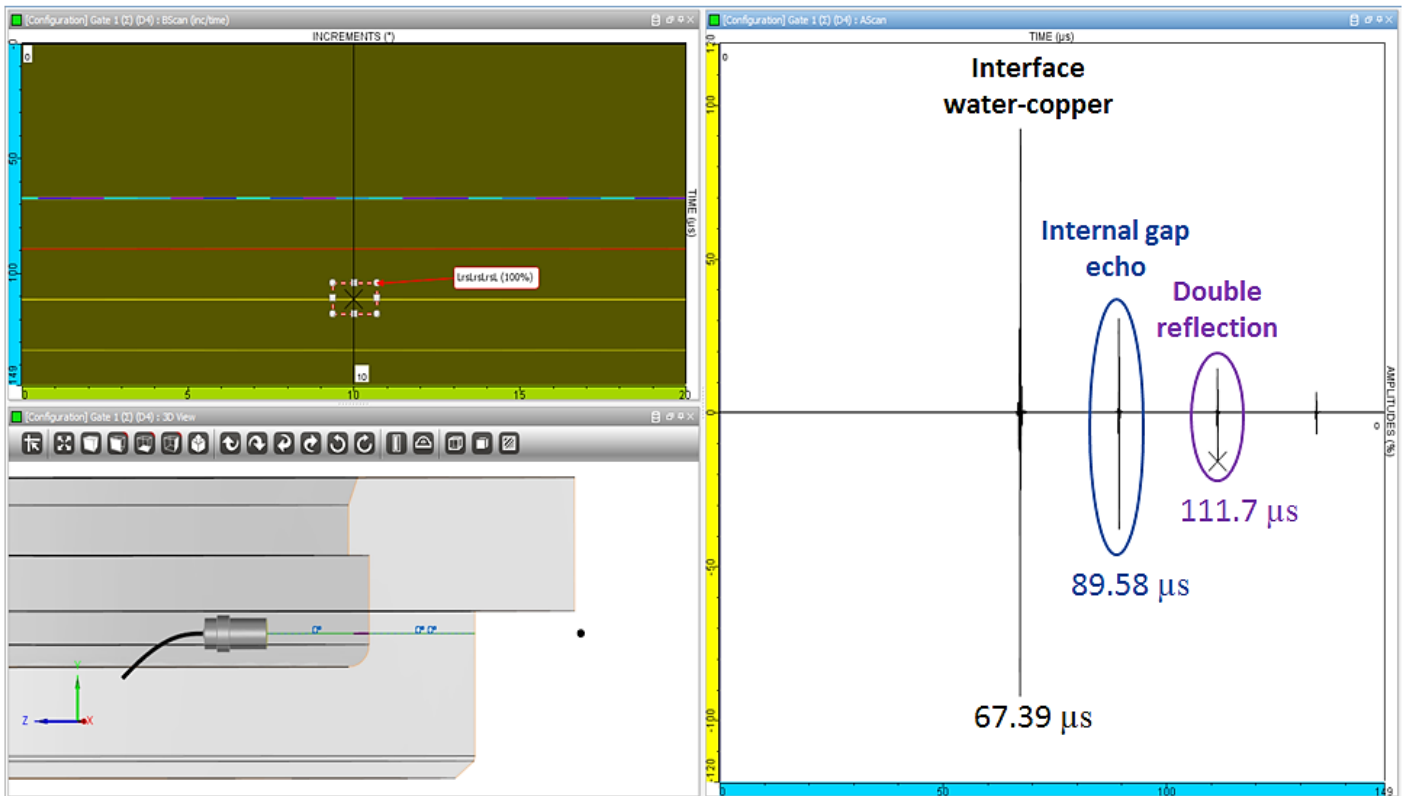


Figure 10. Inspection simulation implemented by CIVA software for the verification of the time of flight of the double reflection on the internal gap.

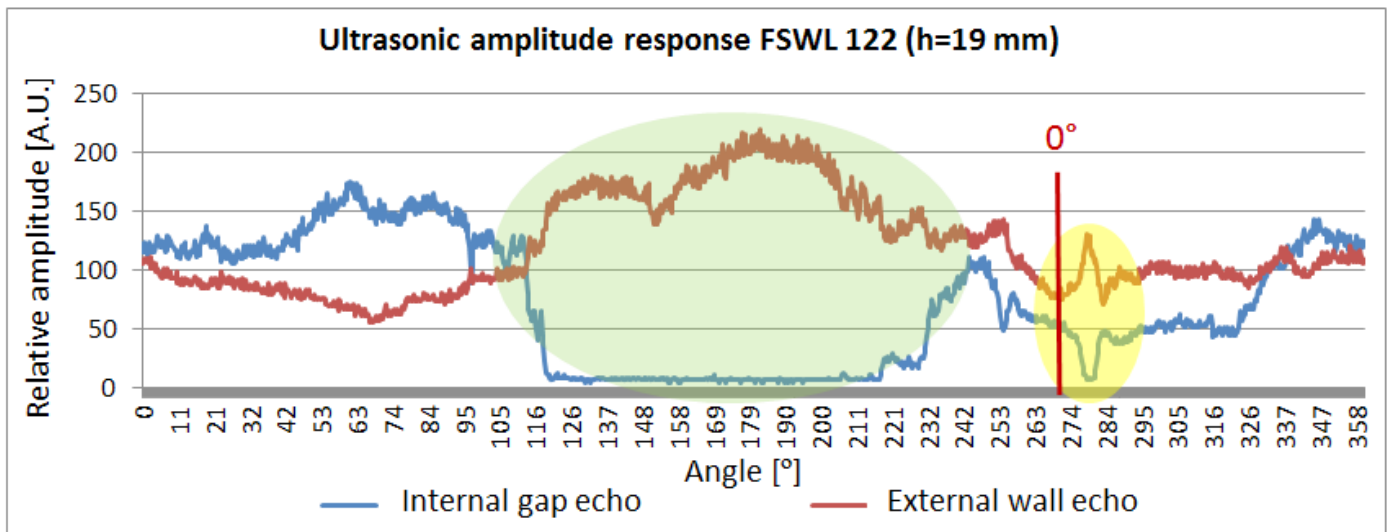


Figure 11. Ultrasonic amplitude response of FSWL 122 at 19 mm height.

The probe is then moved at height 17 mm, 19 mm and 21 mm. The next picture (Figure 11) shows the ultrasonic investigation of the lid at 19 mm of height. This time in blue is reported the amplitude response of gate 2 due to the internal gap reflection and in red the amplitude response of gate 3 that corresponds to the external wall echo. In fact, when the weld joint is perfect, ultrasounds can penetrate in the material without reflection up to the last interface that is the external wall of the canister. As shown in the chart, when the external wall gives a high echo, the internal gap echo is practically zero (green area). The inversion of signals interests an arc of about 100°.

It is interesting to underline the presence of another signal inversion, highlighted in yellow, which could correspond to the point where the welding tool passes two times during the welding process. According to Figure 12, in fact, only when all the parameters are stable, the tool descends at the welding line level and starts the weld (point 4). After 360°, it passes by the starting point and then return back (point 5). This hypothesis is also in accordance with the position of the entry point (vertical red line in the chart) of the welding tool.

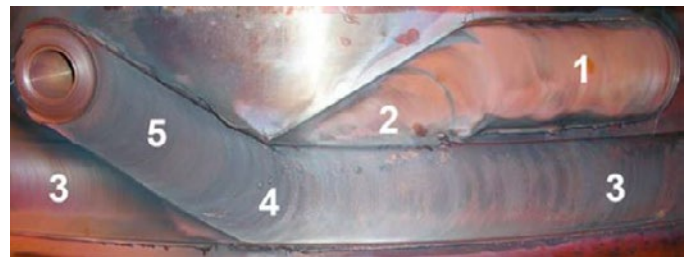


Figure 12. Welding tool rotation and sequences in the welding cycle: 1) acceleration sequence, 2) downward sequence, 3) joint line welding, 4) overlap sequence and 5) parking sequence.

The ultrasonic investigation of the second sample, FSWL 121, generated similar results. In particular Figure 13 reports the internal gap echo and the external wall echo received by the transducer placed at 21 mm of height. As the previous case, an inversion of signals is clearly detectable (yellow area) and probably represents the crossing point of the welding tool (point 4 in Figure 12).

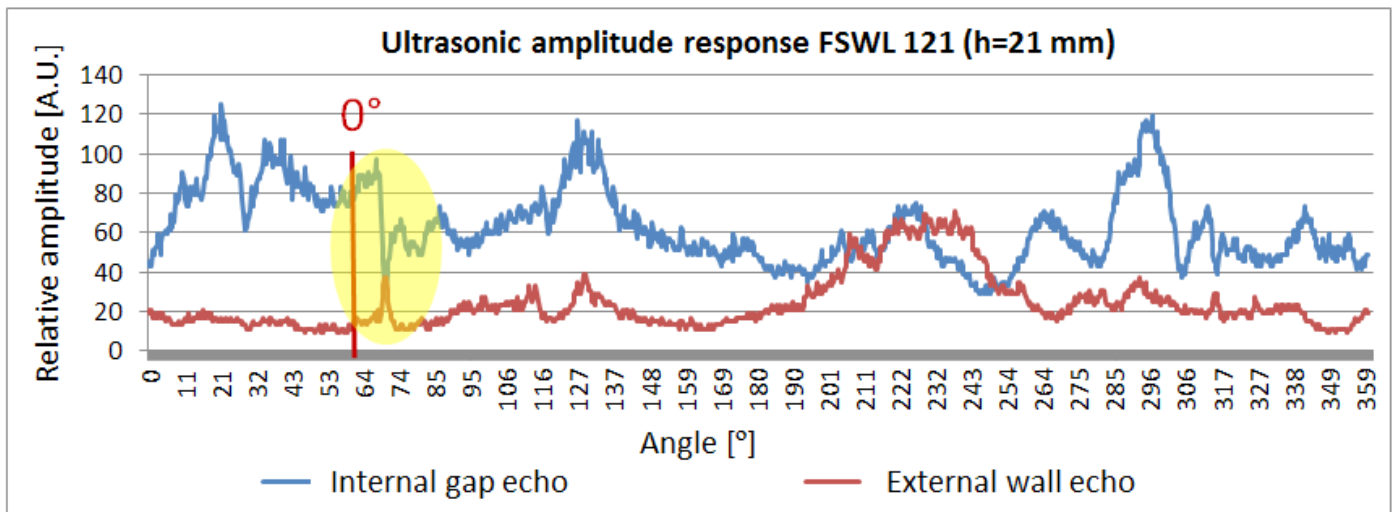


Figure 13. Ultrasonic amplitude response of FSWL 121 at 21 mm height.

	FSWL 121				FSWL 122			
	h=15mm	h=17mm	h=19mm	h=21mm	h=15mm	h=17mm	h=19mm	h=21mm
Correlation indexes between different acquisitions	0.96	0.80	0.65	0.66	0.80	0.92	0.96	0.81
	0.97	0.79	0.80	0.75	0.64	0.92	0.88	0.72
	0.95	0.82	0.62	0.47	0.89	0.97	0.87	0.78
	0.95	0.83	0.99	0.79	0.88	0.97	0.83	0.81
	0.96	0.98	0.75	0.67	0.93	0.95	1.00	0.92
	0.96	0.80	0.86	0.68	0.83	0.91	0.97	0.84
	0.98	0.61	0.63	0.47	0.71	0.90	0.89	0.97
Average value	0.96	0.80	0.76	0.64	0.81	0.93	0.91	0.84

Table 1. Correlation indexes between acquisitions at 15-17-19-21 mm of height.

Following the analysis of echoes at different heights, fingerprints are compared to verify the repeatability of measures. The Pearson’s correlation coefficient is calculated between acquisitions carried out at the same height. The resulting values of correlation indexes are collected in Table 1, and the average value for each height is evaluated. Results put in evidence that the IDA reader is able to acquire ultrasonic signals with a good precision because correlation indexes are higher than 0.8 in most of the cases. However for FSWL 121, measurements acquired at 19 mm and 21 mm present lower values, probably due to an incorrect fixing of the rod holding the transducer.

#### 4. Conclusions

The identification and authentication of copper canisters is of utmost importance to guarantee the Continuity of Knowledge of spent nuclear fuel during transport, storage and deposition in the geological repository. In the past, different technologies were assessed for the labelling of spent fuel casks and, among them, ultrasonic systems would be the best option [13]. The SILab of the Joint Research Centre in Ispra developed an ultrasonic method for copper canister identification and authentication. The identification could be

realized by the machining of chamfers on the inner surface of the lid, forming a unique code readable by an ultrasonic transducer immersed in water, rotating around the lid circumference. The authentication, instead, could be verified by the ultrasonic inspection of the welding area between lid and tube after Friction Stir Welding. Following the description of the authentication method, this paper reports the experimental tests carried out at the SKB’s Canister Laboratory in Oskarshamn. For this purpose, a new reader prototype called IDA reader is developed. The fingerprints acquired during the visit in Sweden are then analysed and compared. The analysis of signals revealed a quite marked variation of the amplitude of the internal gap echo. The uniqueness of fingerprints could not be verified because of the differences in copper lids geometries; however several featured points were identified on fingerprints. Moreover, the repeatability of fingerprints is proved by making different tests on the same samples. Nevertheless, additional inspections on final welded copper samples with the selected geometry of the lid must be carried out in order to verify uniqueness and identify distinctive features on welding areas. Meanwhile, the ultrasonic method will be presented to international safeguards authorities which would give their feedback on the proposed solution.

## 5. Acknowledgements

We thank our colleagues from the Swedish Nuclear Fuel and Waste Management Co (SKB) who provided us copper samples for experimental tests and gave us support during the visit at the Canister Laboratory.

## 6. References

- [1] "Final Repository for spent nuclear fuel". Available online (accessed on 10/01/2018): <https://www.stralsakerhetsmyndigheten.se/en/areas/radioactive-waste/spent-nuclear-fuel-repository/final-repository-for-spent-nuclear-fuel/>.
- [2] Benz JM., Tanner JE, Smartt HA, MacDougall MR, "Maintaining Continuity of Knowledge (CoK) of 45 Spent Fuel Pools: Tool Survey", Pacific Northwest National Laboratory, Richland, Washington, January 46 2016.
- [3] Hildingsson, L., Andersson, C., Fagerholm, R., "Safeguards aspects regarding a geological repository in 40 Sweden", Swedish Radiation Safety Authority, Stockholm, Sweden, IAEA Safeguards Symposium, 2014, 41 paper CN-220-166.
- [4] Clementi C., Littmann F., Capineri L., "Comparison of Tagging Technologies for Safeguards of Copper Canisters for Nuclear Spent Fuel", *Sensors* 2018, 18(4), 929, March 2018. Available online (accessed on 23/04/2018): <https://doi.org/10.3390/s18040929>
- [5] Chernikova D., Axell K., Nordlund A., Wirdelius H., "Novel passive and active tungsten-based identifiers for maintaining the continuity of knowledge of spent nuclear fuel copper canisters", *Annals of Nuclear Energy*, Volume 75, January 2015, Pages 219-227, ISSN 0306-4549.
- [6] Chiaramello M., Sironi M., Littmann F., Schwalbach P., Kravtchenko V., "JRC Candu sealing 75 systems for Cernavoda (Romania) and upcoming developments", *ESARDA Bull.*, vol. 44, pp. 2939, Jun. 76 2010.
- [7] Clementi C., Calzolari M., Capineri L., Littmann F., "Ultrasonic identification of copper canisters to be used for long term geological repository", *IEEE International Ultrasonics Symposium 2016*, Tours, France.
- [8] Clementi C., Littmann F., Capineri L., Andersson C., Ronneteg U., "Ultrasonic Identification Methods of Copper Canisters for Final Geological Repository", *ESARDA Bulletin*, No. 54, June 2017, pp. 75-81.
- [9] "Design, production and initial state of the canister" TR-10-14 SKB Technical Report, Sweden, December 2010.
- [10] "Corrosion calculations report for the safety assessment SR-Site", TR-10-66 SKB Technical Report, Swedish Nuclear Fuel and Waste Management Co, 2010.
- [11] Savolainen, K., "Friction Stir Weldability of Copper Alloys", M.Sc. Thesis., Helsinki University of Technology, Laboratory of Engineering Materials., 2004, 115 p + 3 Appendices.
- [12] Andrews R E, "Friction stir welding – an alternative method for sealing nuclear waste storage canisters", TWI Ltd, TR-04-16, Cambridge, United Kingdom, December 2004.
- [13] Clementi C., Littmann F., Capineri L., "Comparison of tagging technologies for the safeguard of copper canisters for nuclear spent fuel", *Sensors*, January 2018 (under peer review).



# Usability of Monitoring at the Olkiluoto Repository Site for Safeguards

E. Pentti<sup>1</sup> and O. Okko<sup>2</sup>

<sup>1</sup>Pöyry Finland Ltd, Geoscientific Consulting  
Jaakonkatu 3, P.O. Box 50, FI-01621 Vantaa, Finland

<sup>2</sup>Radiation and Nuclear Safety Authority, Nuclear Materials Safeguards  
Laippatie 4, P.O. Box 14, FI-00811 Helsinki, Finland  
E-mail: elias.pentti@poyry.com, olli.okko@stuk.fi

## Abstract:

*At the Olkiluoto repository site, the operator Posiva Oy runs a multidisciplinary monitoring programme targeted at studying environmental impact, improving the understanding of the natural properties of the site, verifying favourable conditions for long-term safety, and developing methods for monitoring the performance of engineered barriers. The usability of the data produced by the monitoring programme for the implementation of nuclear safeguards is assessed, primarily to detect the excavation of any undeclared underground premises.*

*Microseismic monitoring is currently the only method whose results, located seismic events in Olkiluoto and surroundings, are already used in implementing national safeguards. It is concluded that automatic hydraulic head measurements in deep drillholes and land use monitoring also produce relevant data and findings for safeguards purposes.*

*Hydraulic head is monitored in several drillholes that penetrate the rock volume where the repository will be excavated. These holes are divided into sections, so that head can be measured separately at different depths. The monitored sections are often situated in hydrogeological zones, where fractures in the crystalline bedrock allow groundwater to flow significantly more freely than elsewhere. In some of these zones, a groundwater leak into a new tunnel or drillhole gives rise to a significant decrease of hydraulic head at such a large distance that it can be readily detected in several monitoring sections.*

*Monitoring of land use is based on aerial photographs and maintaining a land use record. These sources are used to regularly update a land use grid covering the whole of Olkiluoto. The aerial photographs and land use grid can supplement other imagery used to verify the declaration of surface constructions.*

*The inclusion of the results of hydraulic head and land use monitoring in the input for the implementation of national safeguards could apparently be achieved by examining material and reports that Posiva already delivers for other purposes. The IAEA can use these reports as open source information.*

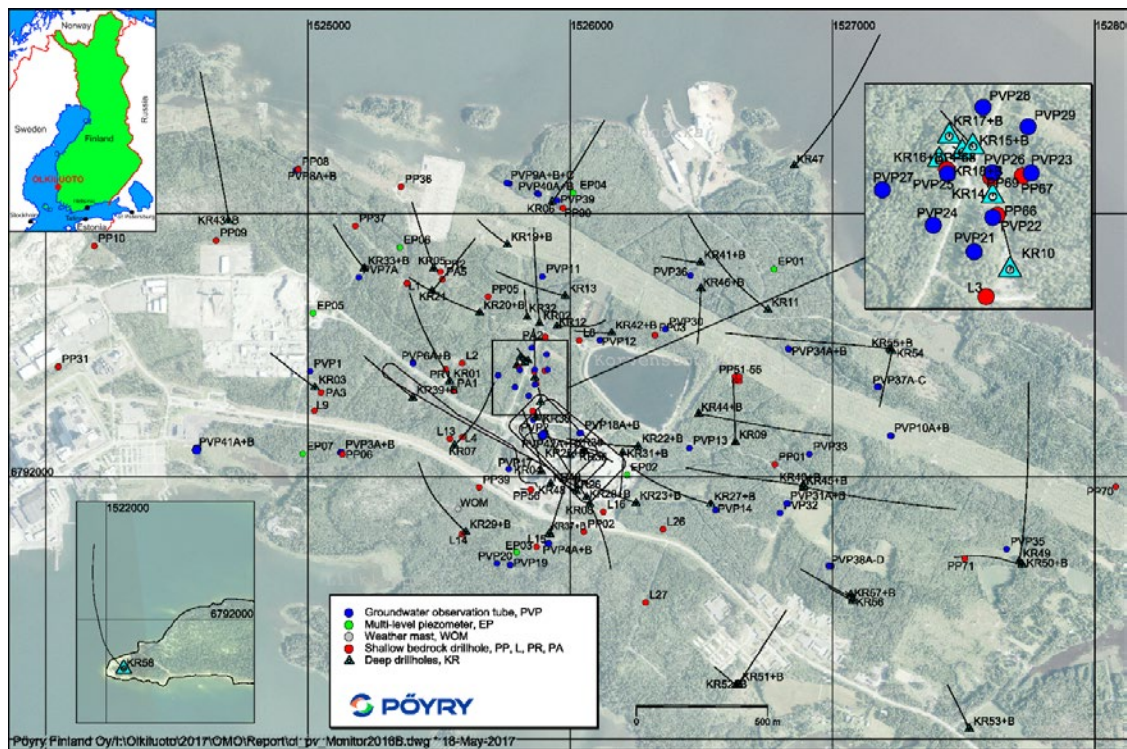
**Keywords:** monitoring; repository; safeguards; seismicity; hydrogeology

## 1. Introduction

Monitoring is required to be performed at a repository as recommended in IAEA TECDOC 1208 [1], and required by STUK Regulatory Guide YVL D.5 [2]. Therefore, Posiva Oy, the company responsible of the final deposition project for the spent nuclear fuel in Finland, has been running an extensive multi-disciplinary monitoring programme at the Olkiluoto repository site. The aims include studying the impact for the repository project on the environment, improving the understanding of the conditions at the site, and supporting the analysis of the long-term safety of the repository.

Before applying for the nuclear construction licence Posiva has constructed an underground rock characterization facility called the ONKALO at the Olkiluoto repository site. It extends to the planned repository depth of about 430 m and includes an access ramp, shafts and technical underground rooms that will eventually also serve as the actual repository. Figure 1 shows the repository site of Olkiluoto, the horizontal extent of the ONKALO in 2017, drillholes and other monitoring points, and in an insert, the location of the site in Finland.

In this article, we discuss the usability of the monitoring programme for implementing national nuclear safeguards at the Olkiluoto site during the construction phase and, as regards detecting undeclared excavation, also during the operational phase. The implementation of nuclear safeguards in an underground repository for spent nuclear fuel mainly concerns the verification of two issues: first, that the construction of underground facilities corresponds to the reported, declared and licenced design, and second, that full accountability for all nuclear material is maintained in the process of transport, encapsulation and final deposition of spent nuclear fuel or any other nuclear material. Of these two issues, the monitoring programme mainly contributes to the first one, because the surveillance of the operation of the facility is not within its scope. The international safeguards requires the declaration of the Design Information, i.e., the layout of the site and the fuel transfer routes per Safeguards Agreement, and according to the Additional Protocol also the buildings, i.e., volumes of underground rooms, but not the monitoring of the stability of the buildings, rooms or premises. These issues have been addressed by the IAEA Expert's Group SAGOR / ASTOR when developing generic safeguards approaches in the



**Figure 1.** Aerial photograph of the Olkiluoto repository site with drillholes and other monitoring points. The surface projections of deep “KR” drillholes are shown as black curves. Grid size 1 km. Insert in the upper left-hand-side corner presents the location of Olkiluoto in Finland.

SAGOR phase [3] and identifying potential technologies for safeguarding geological repositories [4]. The safeguards-safety interface was indicated already by the SAGOR group [5], but the use of operator’s data and safety analysis instead of independent verification has been an obstacle for the IAEA to apply these methods and data available at the repository site.

The operator Posiva runs a safeguards programme since the early stages of the site investigations and the excavation of the ONKALO. Under that programme, the design information is generated e.g. by laser-scanning and maintained in the safeguards-by-design process, and site declarations are updated for the IAEA verification. In addition, detected microseismic events in Olkiluoto have been regularly reported to STUK as the contribution of the monitoring programme to national safeguards implementation. The monitoring results can be used to generate state findings to be ascertained by the IAEA according to the Safeguards Agreement, and moreover, the public results can be included in the IAEA data analytics and included in the state-level evaluation on the fuel cycle-related activities.

The aim of this study is not to develop a independent verification methods for the IAEA, but to facilitate its and in particular STUK’s safeguards assessment by increasing Olkiluoto site understanding using all information available as proposed already in 2006 [6]. The geoscientific monitoring programme at the Olkiluoto repository site was

updated in 2016. Therefore the reassessment of its safeguards relevance was carried out in 2017 [7].

## 2. Monitoring programme and its potential in safeguards implementation

Olkiluoto Monitoring Programme (in Finnish: Olkiluodon monitorintiohjelma, OMO) has formally existed since 2004 [8], when Posiva started the excavation of the ONKALO, although some of the measurements were started more than a decade earlier. The programme has gradually evolved over time on the basis of experience gathered and changes in the needs for research. An updated programme was introduced in 2012 [9] and in 2016, some further adjustments were made and the duration of the programme extended to include the years 2017–2019 by publishing separate updating memos for the six sub-programmes or disciplines: rock mechanics [10], hydrology and hydrogeology [11], hydrogeochemistry [12], surface environment [13], engineered barrier system [14], and foreign materials [15].

### 2.1 Rock mechanics monitoring

Rock mechanics monitoring concentrates on the assessment of tectonic movement and bedrock stability in Olkiluoto and the surrounding area. For the most recent annual monitoring report on rock mechanics, see Haapalehto et al. [16]. Table 1 presents the two methods in the programme that are assessed relevant for the implementation

Process	Method	Location	Frequency	Relevance to safeguards
Seismicity, reactivation of bedrock structures	Microseismic monitoring	18 automatic stations	Continuous	Located seismic events indicate excavation by blasting
Thermal evolution	Monitoring of temperature	Temperature profiles in drillholes	During geophysical and flow loggings	Anomaly in temperature profile may indicate open space near the drillhole

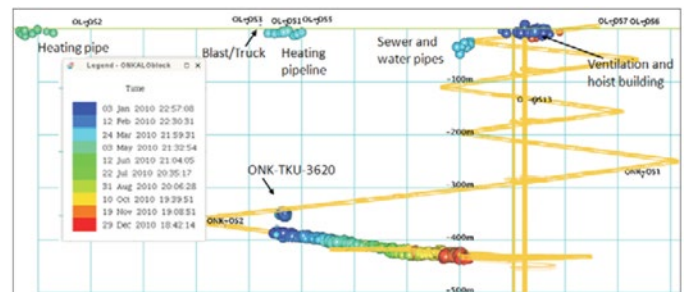
**Table 1.** Targets of rock mechanics monitoring assessed relevant for safeguards.

of safeguards. The first one, microseismic monitoring, is currently the only part of the monitoring programme whose results Posiva submits for safeguard purposes. In addition to the methods in the table, rock mechanics monitoring includes a number of studies that are not considered relevant for safeguards. The tectonic movement of bedrock is monitored by GPS measurements of the relative positions of fixed pillars, and the post-glacial isostatic uplift by precise levelling. In the underground premises, the stability of the excavated rock is monitored by visual observation of spalling and by using extensometry to investigate rock stress redistribution in newly excavated spaces and the possible reactivation of bedrock structures at fracture zone intersections. Tunnel air temperature is also continuously monitored.

Microseismic monitoring is actually aimed at studying natural seismicity and detecting any activation of bedrock fractures that the construction of the repository may induce. However, the bulk of the recorded events are blasts from excavation. The events can be located with sufficient spatial accuracy to ensure that they are related to the licenced construction. As an example of microseismic monitoring data accumulated during one year, Figure 2 presents the seismic events detected in 2010 within the seismic “ONKALO block”, a 2 km × 2 km × 2 km cube surrounding the repository. Most of the events were blasts related to the excavation of the lowest straight section of the access ramp; the marks are coloured on the basis of time, so that the progress of excavation is clearly visible. There also occurred seismic events on or near the ground surface that were associated with construction of pipelines and buildings.

Experience from the time of the excavation of ONKALO has proven that microseismic monitoring is able to detect tunnelling by blasting reliably and accurately. Sensitivity to excavation by boring has also been demonstrated by Saari and Malm [17], as well as the ability to distinguish simultaneous blasting at an undeclared location from declared excavation. The obvious advantages of microseismic monitoring in detecting clandestine tunnelling are that, firstly, it covers the entire volume of host rock between and beyond the network of drillholes and other monitoring locations, and secondly, that blasts are detected immediately. On the

other hand, because of the large sampling frequency of seismic sensors, the measurement data cannot be stored as a continuous time series, but the measuring stations are programmed to store and transmit only the sequences of data where a seismic event occurs according to certain triggering criteria.



**Figure 2.** Microseismic events detected near the ONKALO in 2010 [13]. Grid size is 100 m × 100 m and the colour scale indicates time of the event from January (blue) to December (red). The ONKALO access ramp and shafts are shown in orange.

The second method of rock mechanics monitoring that is assessed to have relevance to safeguards is the monitoring of thermal evolution of bedrock by temperature measurements in drillholes. This assessment is based on the observation of Johansson et al. [19] that the excavation of the tunnel has affected the temperature profiles measured in a deep characterisation drillhole. In the temperature profile acquired in 2015, there are two clearly observable anomalies at depths where the access ramp passes the drillhole at distances of about 20 and 35 metres. The excavation reached these closest points in March 2009 and May 2010, so that the observed temperature effect has taken 5–6 years to develop. Although this demonstrates that thermal monitoring can potentially detect unknown tunnels, the method is evidently very slow and uncertain for the following reasons: the tunnel has to pass a drillhole relatively closely, it takes a long time before the existence of the tunnel alters the temperature of the surrounding rock mass sufficiently for detection (depends on distance but typically order of years), and temperature profile measurements are not carried out systematically in all drillholes but only in those that are selected, by other criteria, for groundwater flow logging.



Process	Method	Location	Frequency	Relevance to safeguards
Evolution of hydraulic head	Hydraulic head monitoring	Packed-off surface drillholes and ONKALO drillholes	Hourly	Detects tunnel excavation in case it causes a change in the flow of groundwater from a monitored hydrogeological structure.
	Analysis of pressure responses	Hydraulic head data	During geophysical and flow loggings	

**Table 2.** Targets of hydrogeological monitoring assessed relevant for safeguards.

## 2.2 Hydrological and hydrogeological monitoring

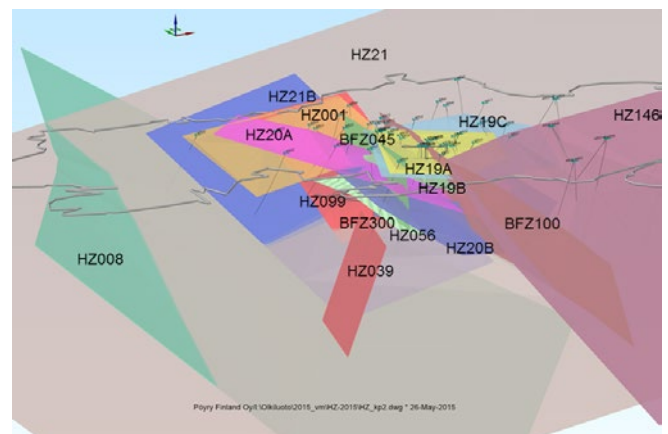
Hydrological and hydrogeological monitoring comprises of studies of groundwater level, hydraulic properties of the bedrock and overburden, hydraulic head and flow of groundwater in the bedrock, inflow into tunnels, and the influence of the Korvensuo Reservoir, the only remarkable body of surface water in Olkiluoto. For the most recent annual monitoring report on hydrology and hydrogeology, see Vaittinen et al. [20]. Of all the related measurements, only the automatic monitoring of hydraulic head of groundwater in packed-off deep drillholes, and the analysis of pressure responses in the head data, are assessed relevant for the implementation of safeguards (see Table 2). Hydraulic head is a quantity used in hydrogeology to express groundwater pressure, equal to the elevation of the (real or theoretical) surface of a column of water connected to the groundwater system. It is more practical than the actual pressure because it is the gradient of head, not of pressure, that determines the flow velocity of groundwater.

In addition to hydraulic head monitoring, some other methods of hydrogeological monitoring can also yield indications of excavation or construction on ground surface, but with such uncertainty and long delay that their relevance to the implementation of safeguards is merely hypothetical. These methods include the monitoring of groundwater level in shallow drillholes and groundwater observation tubes, and the monitoring of groundwater flow and hydraulic properties in deep drillholes by flow logging. In a few cases during the construction of the ONKALO, earthwork on the surface and tunnel excavation have affected groundwater level to an observable extent, but with a delay and at a short range only, so that the activity has evidently been first observed visually.

Hydraulic head is monitored in most of the almost 60 deep (up to a depth of 1 km) characterization drillholes in Olkiluoto. To enable head monitoring and to prevent artificial hydraulic connections in the vertical direction, the drillholes have been packed-off, in other words equipped with a set of inflatable packers that divide the drillhole into hydraulically isolated sections. A maximum of eight packer sections in one drillhole can be connected to the top of the drillhole with a hose so that the water level in the hose and, therefore, the hydraulic head in the section can be measured. Hydraulic head data together with results on groundwater flow and hydraulic conductivity are used to study the effect of excavation on the groundwater system, in

hydrogeological modelling of Olkiluoto, and in the interpretation of hydrogeochemical observations.

The ability of hydraulic head monitoring to detect tunnel excavation and other underground activity results from the repository being constructed in crystalline bedrock, where fracturing and thus also hydraulic conductivity is concentrated in deformation zones that have formed during the geological evolution of Olkiluoto. For a detailed description of the geology of the site, see Aaltonen et al. [21]. The data on the geology and hydrogeology of Olkiluoto, gathered by various methods including monitoring measurements, has been used to compile a hydrogeological structure model of the site [22]. It describes the hydraulic properties of the bedrock with approximately planar hydrogeological zones, along which groundwater is able to flow significantly more easily than in the rock volumes in between. Figure 3 presents a 3D visualisation of the current hydrogeological structure model of the Olkiluoto area. Monitoring of hydraulic head mostly concentrates on the modelled zones, because they are essential for both the planning and the long-term safety analysis of the repository.

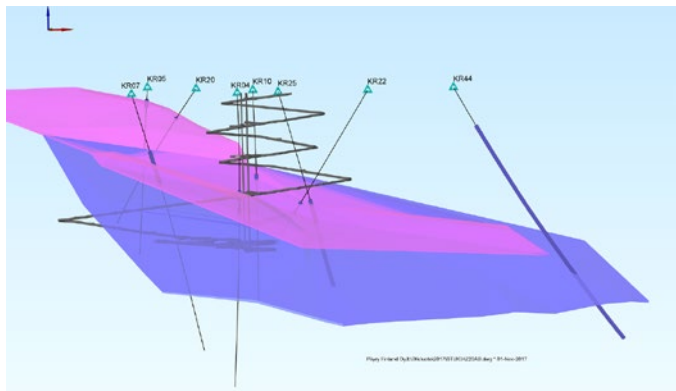


**Figure 3.** Visualisation of the hydrogeological structure model of Olkiluoto. Coloured polygons represent hydrogeological zones (HZ+number) or brittle deformation zones (BFZ+number), the thick grey line the shoreline of Olkiluoto, and thin black lines deep drillholes. View from the south-west.

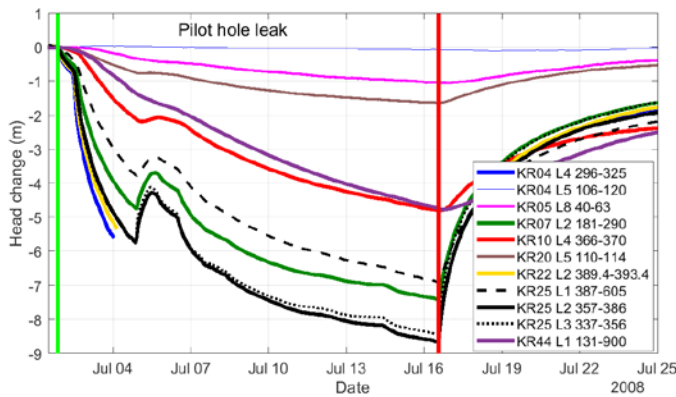
The sensitivity of hydraulic head monitoring to excavation is most clearly demonstrated by data from the time when the construction of the ramp reached the hydrogeological HZ20 system, consisting of zones HZ20A and HZ20B of the structure model, in the summer of 2008. Figure 4 shows these zones, the ONKALO in the extent in 2017, and some of the



drillholes with hydraulic head monitoring sections. Before blasting through the zone, a core-drilled pilot hole was made into the planned tunnel profile for investigations. The pilot hole penetrated the HZ20 system, causing a leak that lasted for over two weeks. Figure 5 presents a plot of the change of hydraulic head during the leak in selected monitoring sections of drillholes intersecting the HZ20 system. The largest head response occurred in section L4 of drillhole OL-KR4, which lies only a few dozen metres from the leaking point. The interruption in the data from that section, as well as the almost as strongly affected L2 of OL-KR22, results from the water level in the measuring hoses in the drillholes falling below the pressure sensor. Uninterrupted data exists from section L2 of OL-KR25 (230 m from the leaking point), where the head decreased by about 8.5 m before the leak stopped. In other monitored drillhole sections in the HZ20 zone, the response decreases with distance still being about 1 m in section L8 of OL-KR5, which lies about 900 m to the north-west of the leak point, and 1.6 m in L1 of OL-KR44, 1,000 m to the east.

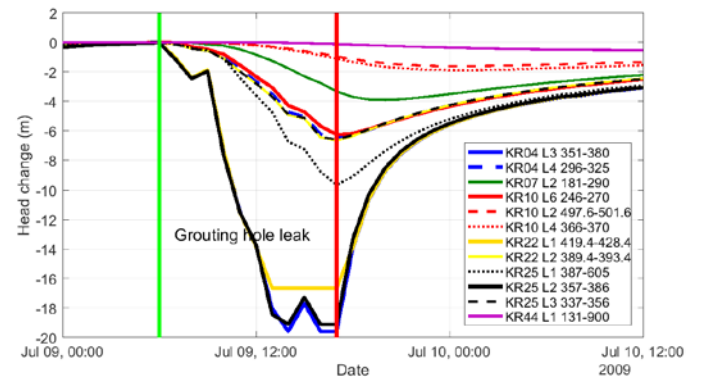


**Figure 4.** 3D illustration of the ONKALO (grey), hydrogeological zones HZ20A (purple) and HZ20B (blue), and some head monitoring sections of drillholes where responses to the leaks discussed in the text were detected. The drillholes are presented with black lines and the selected monitoring sections with thick blue lines. The spiralling access tunnel is about 5 km long and reaches the depth of 450 m from ground surface. View from the south.



**Figure 5.** Change of hydraulic head in some monitored drillhole sections during a leak from a pilot hole intersecting the HZ20 structure. The vertical green and red lines mark the beginning and end of the leak, respectively. Drillhole section labels consist of a drillhole code (KR+number), section code (L+number), and range of drillhole length in metres.

The second example is also related to the HZ20 system. In July 2009, as a preparation for the raise boring of one of the vertical shafts, grouting holes were drilled at the level of zone HZ20. During a leak from one of the holes, head changes graphed in Figure 6 occurred. In about 12 hours, head decreased by almost 20 m in drillhole sections L3 of OL-KR4, L2 of OL-KR25, and L1 of OL-KR22. The response was much smaller or zero in other sections of the same drillholes, demonstrating how hydraulic effects propagate significantly better along the hydrogeological zones than in other directions.



**Figure 6.** Change of hydraulic head in some monitored drillhole sections during a leak from a shaft grouting hole intersecting the HZ20 structure. The vertical green and red lines mark the beginning and end of the leak, respectively. Drillhole section labels consist of drillhole code (KR+number), section code (L+number), and range of drillhole length in metres.

During the excavation of the ramp, dozens of responses to temporary groundwater leaks, similar to the two examples presented here, have been observed. Most of them have been mediated by zones HZ19 and HZ20. Moreover, in a number of monitored drillhole sections, a long-term drawdown (decrease of head) has developed due to hydraulic connections to the underground premises. On the basis of this experience, hydraulic head monitoring data is sensitive to tunnelling in the repository site. When excavation or drilling intersects a major hydrogeological zone or a local hydraulically conductive feature, groundwater pressure is inevitably affected, and the effect propagates to distances of hundreds of meters in a matter of hours. Advantages in comparison with microseismic monitoring are, firstly, that continuous monitoring data is automatically stored from all operational sensors and, secondly, that the effect of excavation is not instantaneous but usually lasts for at least a couple of days even if the leak itself is quickly stopped. Therefore, missing a signal because of failed triggering of the measurement system is not possible. On the other hand, there are the evident limitations that, firstly, a response can usually only be observed if the tunnel or drillhole penetrates a hydrogeological zone that also intersects monitored drillhole sections, and secondly, the exact location of the leak causing the head decrease cannot be determined from the data because of the heterogeneity of

Motivation	Subject	Target/method	Frequency	Relevance to safeguards
Interaction between surface environment and groundwater in bedrock	Land use	Aerial photographs	Every other year	Can supplement present satellite imaging
		Records of changes in infrastructure and other land use	Continuous	Can supplement present accounting of surface facilities
		Update of land use grid	Next 2018	

**Table 3.** Targets of monitoring of surface environment assessed relevant for safeguards.

the structures mediating the effect. However, a rough estimate of the location can be deduced if the same effect is observed in more than one monitoring section.

### 2.3 Hydrogeochemical monitoring

Hydrogeochemical monitoring studies the evolution of groundwater properties and salinity distribution both in the overburden and deep in the bedrock. The principal method is taking and analysing groundwater samples from various targets. Some simple chemical parameters are also monitored continuously in situ, for example pH and conductivity of groundwater leaking into the tunnel. Issues of interest range from the natural chemical and microbiological properties of groundwater in the repository site to human influence due to foreign materials used underground. For the most recent annual monitoring report on hydrogeochemistry, see Lamminmäki et al. [23].

It is, in principle, conceivable that undeclared tunnel excavation or construction on the surface would give rise to detectable changes in groundwater chemistry by disturbing groundwater flow and introducing foreign substances. However, such effects are likely to be slow, limited in range, and ambiguous to interpret. The relevance of all geochemical monitoring to the implementation of safeguards is thus hypothetical at best.

### 2.4 Monitoring of the surface environment

Monitoring of the surface environment includes long-term investigations to acquire site-specific input data for biosphere modelling, research of the interaction between surface environment and groundwater, and studies of the environmental impact of the final disposal project. Moreover, radiological studies aimed at establishing a baseline for the future monitoring of radioactive releases from the disposal facility have been part of the programme, but in the 2016 update, they were organized into a separate project. For the most recent annual monitoring report on surface environment, see Pere et al. [24].

Among the studies of surface environment, the monitoring of land use is assessed relevant to the implementation of safeguards (see Table 3). It involves aerial photographs taken every other year, keeping record of changes in infrastructure and other land use, and maintaining a land use grid describing the principal use of every 50 m × 50 m square of Olkiluoto. All these data are useful for

supplementing the present material used to verify Posiva's design information and site declaration.

Some targets of the monitoring of surface environment have hypothetical but no practical relevance to safeguards: noise measurements or the chemical monitoring of a sedimentation pool containing process water pumped from the underground premises and of ditches that lead waters from the construction site and rock spoil piling area to the sea could, in principle, reveal undeclared activity, but similarly with the hydrogeochemical monitoring discussed above, with high uncertainty and in an ambiguous way. The rest of the targets, like studies on the quality of sea and drainage water, recording forest and aquatic management activities, monitoring surface hydrology and meteorology, and evaluating the impact on exploitable natural resources, have no relevance to safeguards.

### 2.5 Monitoring of the engineered barriers

In the KBS-3V [25] final deposition concept that Posiva plans to implement, the spent nuclear fuel is encapsulated in the original fuel elements into cylindrical canisters with a copper casing surrounding a cast iron interior. After emplacement into vertical deposition holes, the canisters are surrounded with a buffer of bentonite clay blocks, and finally the tunnels are backfilled with bentonite, and tunnel openings and drillholes are closed with various plugs and seals. The canister, bentonite buffer, and tunnel backfill constitute the "engineered barrier system" (EBS) that together with the natural barrier of bedrock is intended to ensure containment of the deposited radioactive material, protection against external disturbances, and retention and retardation of any releases. Posiva's monitoring programme includes a separate discipline for EBS monitoring, which is still in the development stage. Therefore, EBS monitoring does not currently produce results relevant for the implementation of nuclear safeguards.

## 3. Summary and conclusions

This article discusses the Olkiluoto Monitoring Programme and its potential in implementing nuclear safeguards on the disposal facility for spent nuclear fuel that Posiva Oy is constructing in Olkiluoto, Finland. A systematic assessment of each monitoring method leads to the conclusion that three of them produce safeguards-relevant results: microseismic monitoring, automatic hydraulic head monitoring in deep

drillholes, and land use monitoring. In addition, some methods can, in principle, indicate surface excavation or tunnelling, but only at a short distance (if at all) and after the activity would already have been detected visually.

Results of microseismic monitoring, i.e. the detected and located seismic events in Olkiluoto and the surrounding region are reported to the Finnish Radiation and Nuclear Safety Authority (STUK) for safeguards assessment since the early stages of the excavation of the repository. This method of monitoring has proven to accurately detect blasts from underground excavation as well as on the surface.

Automatic hydraulic head (groundwater pressure) monitoring acquires hourly data from over 200 packer sections of deep drillholes in Olkiluoto. A significant share of the monitored sections have been positioned in sub-horizontal hydrogeological zones, where pressure variations, caused by groundwater leaking from the zone into drilled holes or excavated spaces, have been observed to spread over long distances. Therefore, hydraulic head monitoring, has potential to reveal clandestine tunnelling or drilling from the ground surface towards the depth of the disposal facility.

The advantages of the hydraulic head monitoring include sensitivity to all methods of excavation in contrast to microseismic monitoring that can reliably only detect blasting. Moreover, the effects on head that can reveal underground activity are long-lasting or even irreversible, so the probability of missing a significant signal is low. The most obvious disadvantages are that the source of the signal cannot be located with the same accuracy as in microseismic monitoring, and that the method is sensitive only to activities within the hydraulically conductive zones. Posiva already reports interpreted results of hydraulic head monitoring regularly for the supervision of the construction and long-term safety of the disposal facility. Thus, this information could with relative ease be taken into account in the implementation of nuclear safeguards.

The monitoring of land use in Olkiluoto involves aerial photography and updating a land use grid every second year. These results, if reported to STUK for safeguards purposes, can be used to supplement other aerial or satellite imagery of the Olkiluoto site in verifying the declared surface constructions and activities. The monitoring reports are published and thus these can be used by the IAEA as open source information when analysing the nuclear fuel cycle-related activities in Finland.

## References

- [1] *Monitoring of Geological Repositories for High Level Radioactive Waste*; IAEA-TECDOC-1208. IAEA; Vienna, Austria; 2001.
- [2] *Guide YVL D.5: Disposal of Nuclear Waste*; Radiation and Nuclear Safety Authority; Helsinki, Finland; 2013.
- [3] *Safeguards for the Final Disposal of Spent Fuel in Geological Repositories*; STR-312, 5 Volumes, IAEA, Vienna, Austria 1997.
- [4] *Technologies Potentially Useful for Safeguarding Geological Repositories*; ASTOR Group Report 2011-2016, STR-384, Vienna Austria 2017.
- [5] *Report on the Experts Meeting on Interface Issues and Interaction between Safeguards and Radioactive Waste Management in the Context of Geological Repositories*; STR-338, IAEA, Vienna Austria, 2003.
- [6] Okko O & Rautjärvi J; *Evaluation of monitoring methods available for safeguards use at Olkiluoto geological repository*; STUK-YTO-TR-216, 2006.
- [7] Pentti E & Heikkinen, E; *Usability of Olkiluoto Monitoring Programme for Implementing Nuclear Safeguards*; STUK-TR 28, 2017.
- [8] *Programme of Monitoring at Olkiluoto during Construction and Operation of the ONKALO*; Report POSIVA 2003-05. Posiva Oy; Eurajoki, Finland; 2003.
- [9] *Monitoring at Olkiluoto – a Programme for the Period before Repository Operation*; Report POSIVA 2012-01. Posiva Oy; Eurajoki, Finland; 2012.
- [10] Mattila J; *Monitorointiohjelman suunnitelmapäivitys 2016 – Kalliomekaniikka (in Finnish; title in English: updating of the monitoring programme plan – rock mechanics)*; Memo POS-023623. Posiva Oy; Eurajoki, Finland; 2016.
- [11] Sauramo J; *Monitorointiohjelman suunnitelmapäivitys 2016 – Hydrologia ja hydrogeologia (in Finnish; title in English: updating of the monitoring programme plan – hydrology and hydrogeology)*; Memo POS-023622. Posiva Oy; Eurajoki, Finland; 2016.
- [12] Lamminmäki T; *Monitorointiohjelman suunnitelmapäivitys 2016 - Hydrogeokemia (in Finnish; title in English: updating of the monitoring programme plan – hydrogeochemistry)*; Memo POS-023621. Posiva Oy; Eurajoki, Finland; 2016.
- [13] Pere T; *Monitorointiohjelman suunnitelmapäivitys 2016 – Pintaympäristö (in Finnish; title in English: updating of the monitoring programme plan – surface environment)*; Memo POS-023624. Posiva Oy; Eurajoki, Finland; 2016.
- [14] Lahdenperä J; *Monitorointiohjelman suunnitelmapäivitys 2016 – Tekniset vapautumisesteet (in Finnish; title in English: updating of the monitoring programme plan – engineered barriers)*; Memo POS-023625. Posiva Oy; Eurajoki, Finland; 2016.



- [15] Lahdenperä J; *Monitorointiohjelman suunnitelmapäivitys 2016 - Vieraat aineet (in Finnish; title in English: updating of the monitoring programme plan – foreign materials)*; Memo POS-023626. Posiva Oy; Eurajoki, Finland; 2016.
- [16] Haapalehto S, Malm M, Saari J, Lahtinen S & Saaranen V; *Results of Monitoring at Olkiluoto in 2016 – Rock Mechanics*; Posiva Working Report 2017-47. Posiva Oy; Eurajoki, Finland; 2017.
- [17] Saari J & Malm M; *Seismic Monitoring Experiment of Raise Boring in 2014*; Posiva Working Report 2015-02. Posiva Oy; Eurajoki, Finland; 2015.
- [18] Lahti M (ed.) & Siren T; *Results of Monitoring at Olkiluoto in 2010 – Rock Mechanics*; Posiva Working Report 2011-47. Posiva Oy; Eurajoki, Finland; 2011.
- [19] Johansson E, Haapalehto S, Malm M, Saari J, Kallio U, Koivula H, Lahtinen S, Rouhiainen P & Saaranen V; *Results of Monitoring at Olkiluoto in 2015 – Rock Mechanics*; Posiva Working Report 2016-47. Posiva Oy; Eurajoki, Finland; 2017.
- [20] Vaittinen T, Hurmerinta E, Komulainen J, Nummela J, Pentti E, Tammisto E, Turku J, Karvonen T & Sauramo J; *Results of Monitoring at Olkiluoto in 2015 – Hydrology and Hydrogeology*; Posiva Working Report 2016-43. Posiva Oy; Eurajoki, Finland; 2016.
- [21] Aaltonen I (ed.), Engström J, Front K, Gehör S, Kosunen P, Kärki A, Mattila J, Paananen M & Paulamäki S; *Geology of Olkiluoto*; Report POSIVA 2016-16. Posiva Oy; Eurajoki, Finland; 2016.
- [22] Vaittinen T, Ahokas H, Nummela J, Pentti E & Paulamäki S; *Hydrogeological structure model of the Olkiluoto Site in 2015*; Posiva Report (in preparation).
- [23] Lamminmäki T, Pitkänen P, Penttinen T, Pentti E, Komulainen J, Loimula K, Wendling L, Partamies S & Ahokas T; *Results of Monitoring at Olkiluoto in 2015 – Hydrogeochemistry*; Posiva Working Report 2016-44. Posiva Oy; Eurajoki, Finland; 2017.
- [24] Pere T (ed.), Sojakka T, Aro L & Lipping T; *Results of Monitoring at Olkiluoto in 2014 – Environment*; Posiva Working Report 2015-45. Posiva Oy; Eurajoki, Finland; 2017.
- [25] *Long-term safety for the final repository for spent nuclear fuel at Forsmark. Main report of the SR-Site project (updated in May 2015)*; Report SKB TR-11-01. Swedish Nuclear Fuel and Waste Management Co (SKB). 2011.

# Elements of a Swedish Safeguards Policy for a Spent Fuel Disposal System

G. af Ekenstam, L. Hildingsson, R. Fagerholm, C. Andersson

Section for Nuclear Non-proliferation and Security, Swedish Radiation Safety Authority  
Solna Strandväg 96, SE-171 16 Stockholm, Sweden

## Abstract:

*This paper presents an outline of a proposed Swedish encapsulation and deposition system for spent nuclear fuel, possible national measures in support of international safeguards, and possible national measures implemented for domestic purposes. All these measures are only in support of nuclear material accountancy and are not in any way aimed at other scenarios that would be in violation of Swedish law, e.g., theft, falsification, sabotage, etc. Only the operational phase of the geological repository is considered in this paper.*

*The IAEA has developed safeguards approaches under integrated safeguards for encapsulation plants and geological repositories. The approaches are very generic for these two facility types and cannot be used for devising detailed safeguards approaches. In this context, a compatibility evaluation of the generic IAEA approaches vis-à-vis the proposed Swedish system has been conducted. This evaluation also takes into account the conclusion drawn under the Additional Protocol, i.e., the confirmed State-wide absence of undeclared nuclear activities.*

*Two elements of the proposed Swedish system that will need careful consideration are: (1) the high throughput encapsulation process—which may limit the time available for safeguards measurements; and (2) the unavailability of the copper canisters for measurement and evaluation of C/S once they have been loaded into transport casks. While also taking into consideration that ongoing daily operations over a period of several decades is expected at both facilities, there is apparent justification to develop very robust techniques for unattended verification and monitoring involving remote data transition capabilities.*

*For the proposed Swedish system, it appears imperative that the transport casks containing the canisters are covered by robust C/S measures from the time of canister loading at the encapsulation plant up to the time of entering the underground areas of the geological repository. It is considered undesirable to have routine inspection activities (including C/S activities) conducted underground.*

*Lastly, due to safety requirements, the operator is expected to perform comprehensive measurements on all individual fuel elements. These measurement results, in addition to equipment, may also be used by the IAEA. Consequently, authentication and sharing issues may need to be addressed.*

**Keywords:** Final disposal; spent nuclear fuel; safeguards.

## 1. Introduction

Spent nuclear fuel from reactors must be managed and disposed of in a safe manner, including safeguards. A Swedish system for handling spent nuclear fuel has been developed and proposed by the Swedish Nuclear Fuel and Waste Management Company (SKB). In brief, the system is based on encapsulating the spent fuel in copper canisters and depositing them in granite bedrock about 500 m below ground. In 2011 SKB formally submitted licence applications for an encapsulation plant and a final repository.<sup>1</sup>

Spent fuel from Swedish reactors is shipped to Clab, an interim storage facility located in Oskarshamn. Here, the spent fuel is placed in storage pools in the bedrock about 30 m underground. Clab has been in operation since 1985 and is used to store spent fuel from all the nuclear power plants in Sweden.<sup>2</sup> Today there are about 33,000 spent fuel assemblies, at Clab corresponding to 6,500 tonnes of uranium and 61 tonnes of plutonium.<sup>3</sup> The spent fuel stored at Clab consists primarily of BWR and PWR fuel with a few additions of older experimental fuel and spent fuel debris.<sup>4</sup> The flow of spent nuclear fuel in Sweden is illustrated schematically by Figure 1. A proposed encapsulation plant and a geological repository are also included in the figure.

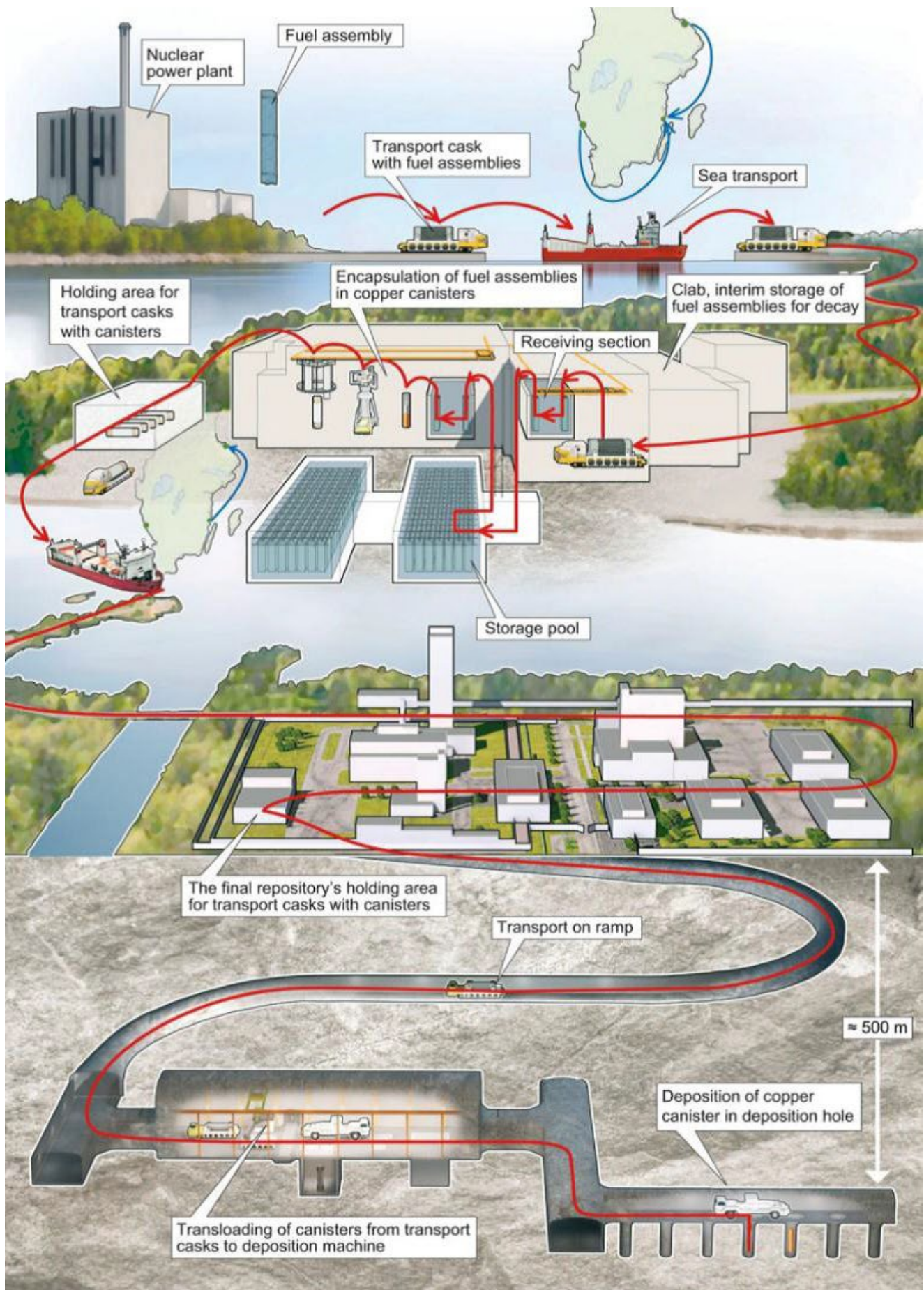
<sup>1</sup> SKB's licence applications under two separate legal instruments have been reviewed by the Swedish Radiation Safety Authority and by the Land and Environmental Court. The Swedish Radiation Safety Authority supports licensing under the Act on Nuclear Activities on condition that a step-wise authorisation process is followed for key future phases of development. The conclusion from the Land and Environment Court is that some identified uncertainties in the long term stability of the copper canisters need to be further addressed by SKB before a licence under the Environmental Code can be considered. It is the Swedish government that takes the final decision.

<sup>2</sup> The NPPs at Forsmark, Oskarshamn, Ringhals and Barsebäck (the latter undergoing decommissioning).

<sup>3</sup> As of March 2017.

<sup>4</sup> This is fuel from the closed, experimental Ågesta reactor and some German fuel obtained in a swap with Swedish fuel that was intended to be reprocessed. The fuel debris consists of parts of spent fuel rods from the Studsvik Hot Cell laboratory. This is debris from examination of the fuel or leaking fuel rods that have been cut in smaller parts. The fuel debris is stored in closed containers.





**Figure 1.** Schematic illustration of the flow of spent fuel in Sweden, from nuclear power plants to final deposition. Source: SKB and LAJ Illustration.



## 2. The encapsulation plant

SKB has applied for permission to build the encapsulation plant, which is to be co-located with the existing interim storage facility Clab as an extension above ground. Thus there will be no need for transports between the interim storage and the encapsulation plant. The combined facility will be named 'Clink'.<sup>5</sup> Cooling times of the spent fuel that will be encapsulated will typically be 40 years, but it may vary from 10 to 60 years. Burn-up will range from a few GWh/tU up to 60 GWh/tU.

Fuel to be encapsulated will be moved to a measuring position where the operator will verify important parameters of the fuel, such as thermal residual power and burn-up. After the operator's verification, the fuel will be moved to a transfer canister, which will be moved to the handling cell where the assemblies will be dried and placed in a copper canister. In a series of steps, a copper lid will be put on and stir welded to the copper canister. The weld will be quality checked by the operator and the surface of the canister will be polished and decontaminated. Lastly, the canister will be placed in a transport cask and temporarily stored at the facility before being shipped to the geological repository site.

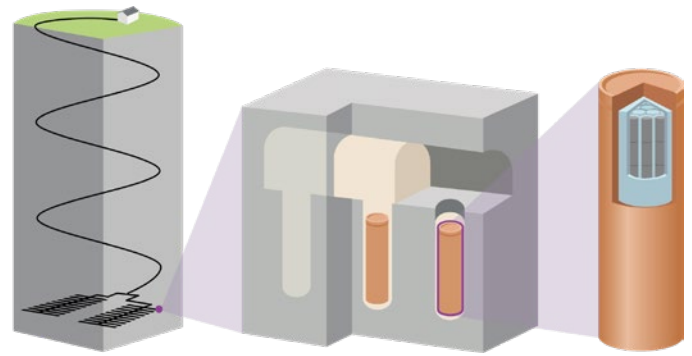
Each copper canister will have an insert of cast iron with positions for 12 BWR fuel assemblies or four PWR fuel assemblies. Fuel will be encapsulated during campaigns arranged separately for BWR and PWR fuel. It is envisaged that 150 canisters will be treated per year. During routine operation, this means loading one canister per workday, corresponding to a flow of 12 BWR assemblies, or four PWR assemblies, per day.

## 3. The geological repository

The plan is to build the geological repository at Forsmark, about 360 km north of the encapsulation plant. The repository will be close to, though separated from, the Forsmark NPP and the final storage facility for low and intermediate level radioactive waste, SFR, located there.

The geological repository will consist of a surface area and an underground deposition part, about 500 m below ground. The surface area will encompass a terminal and buildings for elevators, ventilation and backfill materials. There will be a transport ramp for vehicles connecting the above ground area with the underground repository; this will include the vehicle for transporting the transport casks containing the copper canisters. Copper canisters from the transport vehicle will be reloaded to a deposition machine in the underground Central Area. Transport tunnels will lead from the Central Area to the deposition tunnels,

each having about 30 drilled vertical holes for one copper canister each. When all positions in a deposition tunnel have been filled, the tunnel will be backfilled and sealed with a concrete plug. A schematic illustration of the geological repository site is shown in Figure 2.



**Figure 2.** A schematic illustration of the geological repository's proposed layout.

Deposition tunnels will be excavated in a rock excavation zone, separated from the deposition and backfilling zones by a protection zone (with no blasting) and a separation wall. Excavation, deposition and backfilling can thus take place simultaneously, although physically separated. When the deposition tunnels have been backfilled, the separation wall will be moved, and the next step of excavation, deposition and backfilling can begin. One such step will take at least one year.

A specially designed ship will deliver transport casks containing filled copper canisters from the encapsulation plant to the geological repository. The transport casks will be temporarily stored at surface level in a terminal building before being transported by a ramp vehicle underground to the Central Area. The copper canisters will then be transferred from the transport cask into a radiation shield of the deposition vehicle. The deposition vehicle will bring the copper canister from the Central Area to its final deposition position. Lastly, the ramp vehicle will return to the surface with the empty transport cask. The facility will deposit 150 canisters per year during normal operation. This means an average of one transport cask with copper canisters will be transported each workday from the surface terminal building to the subsurface Central Area and deposited.

Both the encapsulation plant and the geological repository will be in operation for about 45 years. After this period of operations, the surface buildings will be removed and the repository sealed. More details on the proposed encapsulation plant and the geological repository can be found in [1].

## 4. Legal requirements and national policy

One basic national legal requirement is that operators of nuclear facilities are responsible for ensuring that all the necessary measures are taken for safe management and

<sup>5</sup> An acronym for the Swedish term Clab and *inkapslingsanläggning* (Swedish for encapsulation).

disposal of spent nuclear fuel. This includes fulfilling all obligations as prescribed by Sweden's agreements aimed at preventing the proliferation of nuclear weapons [2]. As the licensees have assigned management and disposal of spent fuel to the Swedish Nuclear Fuel and Waste Management Company, SKB, the responsibility rests with SKB.

Insofar as a geological repository is concerned, the main national policy in Sweden with regard to nuclear material accountancy is to provide assurance domestically and internationally that all deposited nuclear fuel is as declared. According to national regulations [3], SKB must ensure that sufficient and correct nuclear material accountancy information and knowledge are in place and available on the part of the spent fuel prior to its deposition. This can be carried out by verifying that the documentation accompanying the nuclear material is complete and correct, for example by using a specially designed "paper trail" (e.g., source and operating documents) verification procedure covering the entire fuel history. In the event of uncertainty, SKB should perform the necessary measurements or analyses. SKB is also required to have a system in place guaranteeing that necessary and correct information about the nuclear material is documented and retained following the material's disposal.

As a consequence of Sweden's international safeguards obligations, all requirements must be met as effectively and efficiently as possible. This should involve the inclusion of design features that further facilitate the implementation of international safeguards. In order to achieve this, early discussions between the parties involved will be necessary. Therefore, early provision of the required documentation is of importance for fostering efficient and cost-effective safeguards.

## 5. Safeguards models

The IAEA model integrated safeguards approach for an encapsulation plant [4] assumes that the encapsulation plant is a separate facility and that the spent fuel will be transferred from an interim storage facility in a transportation cask to an assembly handling cell of the encapsulation facility. The proposed Swedish encapsulation plant will, however, be co-located with the spent fuel interim storage facility and form a combined facility. The encapsulation part of the facility will not have an area for receiving and storing spent fuel transport casks. The spent fuel from the NPPs will be stored in the interim storage area and stored in pools for several years before being moved internally to the encapsulation plant.

The IAEA model integrated safeguards approach for a geological repository [5] assumes a separate facility similar to the proposed Swedish system. However, there are a few differences. The IAEA model assumes that the copper canister can be identified upon receipt at the geological repository and that canister identification can be

performed when a canister is transferred between the above ground area and the geological repository at the entrance of the repository. In the proposed Swedish system, however, the copper canisters will be shielded by a transport cask until they reach the underground central area.

These models [4], [5] assumes that during temporary canister storage above ground, dual C/S systems should be applied. A redundant C/S system is to be applied to the disposal canister during transport from the encapsulation plant to the repository [5]. In this context, we want to stress the importance of having robust C/S systems on the transport cask, e.g., systems that can be fully operated by facility employees while also providing credible assurance for the international community.

However, these model approaches are partly outdated and do not fully reflect the current (not yet finally formulated) policy of the IAEA, the findings of SAGOR I-II or the provisions of the Additional Protocol. They can therefore not be used directly as a basis for any detailed technical preparations by Sweden.

## 6. Safeguards considerations

### 6.1 General

On the basis of, inter alia, IAEA GOV/2002/8 [6], IAEA Model Integrated Safeguards Approaches for Spent Fuel Encapsulation Plants [4] and the IAEA Safeguards Glossary (2001) [7], in the absence of finally issued formulated guidance, it is our understanding that the basic international verification requirements for an encapsulation plant are:

- Yearly verification for "gross defects" (yes/no test whether or not all declared fissile material is missing) with "low detection probability" (20%) for spent fuel elements which are available for measurement and which are "difficult to dismantle"; [4]
- Verification for "partial defects" (at least a yes/no test whether or not 50% of the declared fissile material is missing) for each spent fuel element which is being placed in a copper canister and for yearly verification of spent fuel elements which are available for measurement and which are not "difficult to dismantle"; [4]
- Maintaining "dual C/S" or an equivalent system for spent fuel elements which are not available for measurement. [4]

There is no completely clear definition of the concept "difficult to dismantle". Rod exchange has been performed earlier on both BWR and PWR fuel in the ponds of Swedish nuclear power plants. However, with the absence of the required equipment for dismantlement at the Clab and Clink sites, it is reasonable to assume that all fuel that will be deposited can be classified as "difficult to dismantle".

With the above requirements and assumption, it is expected that the nuclear material at Clink will be verified with low

detection probability for gross defects on an annual basis. Spent fuel will be verified for partial defects immediately prior to encapsulation. After encapsulation, canisters will be placed in transport casts and temporarily stored at the site under dual C/S before shipment.

Thereafter a robust C/S system should be applied to the transport cask. This C/S should be evaluated upon entry into the underground area at the geological repository.

The activities under the Additional Protocol are not fully credited for in the two IAEA approaches mentioned above. The confirmed state-wide absence of undeclared activities should render unnecessary certain proposed monitoring and verification activities. In this context, we would refer to an excerpt from the Minutes of the Experts' Group on Safeguards for Final Disposal of Spent Fuel in Geological Repositories [8] and also the statement from DG to the IAEA Board of Governors in February 2002 [9].

*"The important difference is that under Integrated Safeguards, geophysical methods may not be needed to detect excavations or excavation activities. For this purpose, geophysical tools could be replaced with Complementary Access and information analysis. Ground Penetrating Radar (GPR) may still be required for DIV purposes (i.e. detection of undeclared tunnels, rooms and boreholes, such as any permanent underground equipment and installations)." [8]*

*"The measures of the Model Additional Protocol were never intended to be simply superimposed as a new 'layer' of activity on top of safeguards as implemented under INF-CIRC/153 (Corrected) and earlier strengthening measures. Given the additional assurances provided under an additional protocol, the need to avoid undue burden on States and facility operators, and the need for maximum efficiency in the light of the prevailing resource constraints, the new measures were to be 'integrated' with existing ones." [9]*

Periodic DIVs and CAs under and above ground will provide sufficient assurance of the integrity of the site declarations and the absence of undeclared activities for both areas. The implementation of AP measures in the State will add more information on the nuclear capabilities.

As mentioned earlier, the last verification opportunities for the individual fuel elements exist at the encapsulation plant. The operator is expected to perform comprehensive measurements on all individual fuel elements for safety purposes. The optimal position for the operator's performance of these measurements is as early as possible in the material flow into the encapsulation process. This enables the operator to more easily reject assemblies that for safety or other reasons do not fit into the planned canister.

The IAEA (and the Euratom), on the other hand, presumably prefer to have the verification measurement performed immediately prior to the canister lid being put on and

welding being started. This verification is expected to be performed according to established IAEA criteria and practice, namely, a verification for "partial defects" for the spent fuel element.

Routine inspection activities underground at the final repository are not foreseen; underground activities will be limited to DIV only. Also, see the following recommendation from SAGOR:

*"The recommended safeguards approach is to use item accounting supported by a reliable and comprehensive C/S system above-ground to verify, inter alia, the flow of full casks and overpacks. DIV is recommended as the primary safeguards measure underground. DIV would include geophysical methods." [10]*

## 6.2 Measurements and possible use of operators' results

It is not desirable to have two completely different pieces of measurement equipment and perhaps also two different measurement positions for the required final verification of the spent fuel. This takes up space and will take more time. Also, it must be kept in mind that up to 12 assemblies will be encapsulated on a daily basis. It should be investigated to what extent the operators' measurement results and equipment can be shared with the IAEA (and Euratom). It has to be assured that the operator's measurement results in principle are sufficient for the IAEA (and Euratom). Therefore, the authentication issues must also be addressed to provide the international safeguards with the required opportunities for drawing independent conclusions.

Considering the fact that daily operations are expected to take place over the course of several decades, there appears to be a need to develop unattended verification techniques by means of remote data transmission capabilities. The measurement position needs to be arranged at Clink in coordination with the IAEA (and Euratom).

After measurement at the Clink site, proper C/S measures must be applied to assure Continuity of Knowledge (CoK) from the final measurements until the closure of the copper canister. If needed, as a backup to the C/S measures, a simple unattended quality control immediately prior to the assemblies being placed in the copper canister may also be considered. Such verification could involve reading the fuel identification number, measuring the weight of the assembly, and using a gross gamma detector. After the spent fuel has been placed in the copper canister, additional C/S measures have to be applied until the canister is placed in its final position in the geological repository.

If a method is developed and approved, verification of the copper canister may also be conducted at the encapsulation plant. However, for practical reasons, there are limitations to conducting similar verification on the transport



cask or on the copper canister underground at the final repository during normal operations. In exceptional cases, such verification underground could be performed in order to resolve anomalies.

### 6.3 Continuity of Knowledge

In the proposed Swedish system, it seems imperative that the transport casks containing the canisters are covered by robust C/S measures from canister loading at the encapsulation plant to entering the underground part of the geological repository.

The operational activities are expected to be run continuously for approximately 40-50 years with daily production of one copper canister and shipments on at least a bi-weekly basis, a sealing system that can be attached, also that the same seal can be detached by the operator, would be cost-efficient and enhance an efficient use of resources.

The inner walls of the underground tunnels and shafts define the primary containment of the geological repository. During construction and operation of the repository, there will be an access ramp, ventilation shaft, etc. These should be covered by C/S methods that are able to detect movements of spent fuel down to the deposition location and to detect any removal of nuclear material from the underground part. It is important to verify that a canister enters the underground part of the repository. This enables us to treat the underground part of the geological repository as a black box and there is thus no need for C/S and verification methods underground.

Also, as already discussed earlier, it is not considered desirable to have routine inspections of nuclear material accountancy activities, or verification of seals, etc. performed underground.

If the C/S is lost, specific measures determined by the IAEA will be applied [5]. A unique identifier for each copper canister resolution may contribute to its resolution, but not for routine use. Gamma and neutron measurements on the transport container may also be considered as a measure to resolve inconsistencies and re-establish C/S.

Verification of empty transport containers leaving the underground area is to be performed, e.g., weighing, gamma and neutron measurement.

### 6.4 Design Information Verification

The integrity of the geological repository can be verified during DIV, which may be conducted periodically. The main objectives are to confirm the following: that the excavations are performed as declared, that there are no other undeclared nuclear activities, and that there are no clandestine removal routes or excavations. In this context, Complementary Access both above and below ground,

information from satellite imagery and other open sources' information provide assurance for confirming the absence of clandestine activities at the area of the site. Hence, there is no need to continuously monitor the excavation by using geo-seismic monitoring.

## 7. Conclusions

The conclusions drawn under the Additional Protocol are not properly credited for in the IAEA approaches mentioned above. The confirmed state-wide absence of undeclared activities should render unnecessary certain monitoring and verification activities that have been proposed. Therefore, in this context, some of the facility-specific considerations in the IAEA model may not apply.

The last verification opportunities for individual fuel elements and also for routine verification of the canisters will exist at the encapsulation plant. The final spent fuel verification prior to canister welding at the encapsulation plant is expected to be performed according to established IAEA criteria and practice.

The maximum time available for verification will depend on the material flow. In the proposed Swedish spent fuel disposal system, up to 12 assemblies will be encapsulated in one day, so the measurement times will probably be in the order of minutes. Considering the fact that daily operations over the course of several decades are expected, there is a need to develop unattended verification techniques by means of remote data transmission capabilities.

Due to safety requirements the operator is expected to perform comprehensive measurements on all individual fuel elements. It should be investigated if these measurement results can be shared with the IAEA. Therefore, authentication and sharing issues have to be addressed.

Inspections for DIV purposes are essential to confirm that the repository is constructed as declared and to confirm the absence of any undeclared activities. It is considered undesirable to have other routine and verification activities, including C/S, performed underground.

In the proposed Swedish spent fuel disposal system, it seems imperative that the transport casks containing the canisters are covered by robust C/S measures from the time of canister loading at the encapsulation plant up to the time of entering the underground part of the geological repository. It is considered undesirable to have routine inspection activities (including C/S activities) performed underground.

## 8. Acknowledgements

The authors gratefully acknowledge all those who have reviewed the content and contributed by providing

information for this paper, in particular our colleagues at SSM and the SKB Company.

## 9. Legal matters

### 9.1 Privacy regulations and protection of personal data

The authors agree to allow ESARDA to print our names/contact data/photographs/article in the ESARDA Bulletin/Symposium proceedings, or any other ESARDA publications and, when necessary, for any other purposes connected with ESARDA activities.

### 9.2 Copyright

The authors agree that submission of an article automatically authorises ESARDA to publish the work/article in whole or in part in all ESARDA publications – the bulletin, meeting proceedings, and on the website.

The authors declare that their work/article is original and is not a violation or infringement of any existing copyright.

## 10. References

[1] SKB TR-16-15: RD&D Programme 2016. Programme for research, development and demonstration of

methods for the management and disposal of nuclear waste, September 2016.

- [2] Swedish Act on Nuclear Activities (1984:3),
- [3] SSMFS 2008:3,
- [4] Model Integrated Safeguards Approach for a Spent Fuel Encapsulation Plant, SG-PR-1305 Version, 2010-10-06
- [5] Model Integrated Safeguards Approach for a Geological Repository, SG-PR-1306 Version, 2010-10-06.
- [6] IAEA Model Integrated Safeguards, IAEA GOV/2002/8.
- [7] IAEA Safeguards Glossary 2001, Edition International Nuclear Verification Series No. 3.
- [8] Excerpt from Minutes from the Fourth Meeting of the Experts' Group on Safeguards for Final Disposal of Spent Fuel in Geological Repositories, Gimli, Manitoba, Canada, 24-26 July 2001.
- [9] Report by DG to the IAEA BoG, February 2002.
- [10] SAGOR report 1, September 1998, STR-312, Volume 1, Chapter 3, 2nd para.

# Euratom On-Site Laboratories Refurbishments and Developments

L. Duinslaeger<sup>1</sup>, K. Casteleyn<sup>1</sup>, C.-I. Andor<sup>1</sup>, R. Buda<sup>1</sup>, R. C. Marquez<sup>1</sup>, L. Commin<sup>1</sup>, F. D'Amati<sup>1</sup>, L. Emblico<sup>1</sup>, T. Enright<sup>1</sup>, M. Hild<sup>1</sup>, J. H. Domenech<sup>1</sup>, A. Le Terrier<sup>1</sup>, K. Luetzenkirchen<sup>1</sup>, S. Mialle<sup>1</sup>, S. Morel<sup>1</sup>, A. Muehleisen<sup>1</sup>, A. Nicholl<sup>1</sup>, C. Nourry<sup>1</sup>, D. Poublan<sup>1</sup>, M. Ramos Pascual<sup>1</sup>, G. Rasmussen<sup>1</sup>, A. M. Sanchez Hernandez<sup>1</sup>, F. Sarli<sup>1</sup>, S. Street<sup>1</sup>, G. Suurkuusk<sup>1</sup>, G. Toma<sup>1</sup>, M. Toma<sup>1</sup>, O. S. Valu<sup>1</sup>, P. Van Belle<sup>1</sup>, D. Wojnowski<sup>1</sup>, E. Zuleger<sup>1</sup>, F. Lipcsei<sup>2</sup>, P. Schwalbach<sup>2</sup>, S. Synetos<sup>2</sup>

<sup>1</sup>European Commission, Joint Research Centre, Directorate for Nuclear Safety and Security, P.O. Box 2340, D-76125 Karlsruhe, Germany

<sup>2</sup>European Commission, Directorate General Energy, Directorate Euratom Safeguards, L-2920 Luxembourg

## Abstract:

*The two Euratom On-Site Laboratories located at the reprocessing plants in Sellafield, UK (the OSL) and La Hague, France (the LSS) have been operational for more than 15 years. Both On-Site Laboratories, operated by DG JRC on behalf of DG ENER, are instrumental in providing independent assurance to the European Commission that nuclear material is not diverted from its declared use.*

*Over the years, the laboratories have been in constant evolution, some equipment became outdated and in need of refurbishment or replacement. Renewals in the laboratories are foreseen to continue over the coming years. The larger projects entailed replacement of a thermal ionization mass spectrometer, including decommissioning of its connected glove boxes and installation of a new glove box; renewal of the complete Hybrid K-edge analysis system, including installation of an independent gamma station and refurbishment of two glove boxes. Such tasks include the design, manufacture, purchase, installation and testing of the required equipment both at JRC-Karlsruhe and on site, as well as complying with site procedures and detailed requirements to obtain the operator's permission prior to any non-routine work.*

*Over the years, several items of innovative equipment were developed at JRC-Karlsruhe, all tailored to the specific needs of the On-Site Laboratories and taking into account the operational boundary conditions of industrial nuclear facilities. Examples include infra-red based heater for alpha spectrometry with enhanced safety features; a hotplate using a flexible low power heating element; and a semi-automatic chemical separation unit. The replacement of certain equipment parts (e.g. LEMO connections on glove boxes) also required ingenious solutions.*

**Keywords:** Safeguards; refurbishment; equipment; On-Site Laboratories

## 1. Introduction

In the European Union, irradiated fuel from nuclear power reactors is reprocessed at La Hague in France and Sellafield in the United Kingdom. The four reprocessing plants in these two sites are the largest nuclear facilities within the EU, processing hundreds of tons of nuclear material in a year. Under the Euratom Treaty, celebrating its 60<sup>th</sup>

anniversary in 2017, the European Commission has the duty to assure that nuclear material is only used for declared purposes. The Commission, represented by the Directorate General for Energy (DG ENER), assures itself that the terms of Article 77 of Chapter VII of the Treaty have been complied with. In contrast to the Non-Proliferation Treaty, the Euratom Treaty requires the European Commission to safeguard all civil nuclear material in all EU member states – including the nuclear weapons states.

The considerable amount of fissile material separated per year (including several tonnes of Pu) calls for a stringent system of safeguards measures. The aim of Euratom safeguards is to deter diversion of nuclear material from peaceful use by maximising the chance of early detection. At a broader level, it provides assurance to the international community that the European nuclear industry, the EU member states and the European Union honour their legal duties under the Euratom Treaty and their commitments to the Non-Proliferation Treaty.

A thorough analysis of the options to perform nuclear material accountancy and safeguard nuclear material at these reprocessing plants concluded – in the early 1990s – that sampling of material from the process streams would be required. Transport of the samples to a central Euratom laboratory should be avoided for reasons of cost effectiveness, timeliness and risk reduction [1]. Therefore, laboratories were established on the sites of Sellafield and La Hague that opened in 1999 and 2000, respectively.

The major advantages of On-Site Laboratories are: inspectors receive sample results quickly (timeliness); in case of doubt re-verifications can be done with little delay; measures to make sure samples are authentic are simpler to implement; efficiency and cost effectiveness; waste reduction through return of material to the process stream; and reduced transport needs.

## 2. The Euratom Safeguards On-Site Laboratories [2]

In October 1999, DG ENER, the JRC, and the plant operator started up the first "On-Site Laboratory" (OSL) at the site of the reprocessing plants in Sellafield, to cover principally the activities of the newly constructed THORP plant together with several additional samples from the older Magnox



plant [3]. The OSL was planned with additional capacity to accommodate samples from the fuel fabrication plants in the UK (Sellafield MOX Plant, then under construction, and the LEU plant at Springfields). In June 2000, the second On-Site Laboratory, the “Laboratoire-Sur-Site” (LSS) began operations at the reprocessing site in La Hague [4].

The official start of operations took place after a process of conception, planning, design, development, and construction which lasted about one decade. Analysts of JRC-Karlsruhe have since been operating both On-Site Laboratories on behalf of Euratom Safeguards. Both On-Site Laboratories were integrated into existing or new operator laboratories. The On-Site Laboratories are run under the operator’s site licence and can use supporting services. They have to follow all the site safety, operating, and security rules. Staff of the On-Site Laboratories must be trained and certified under the existing site regulations and must comply with the same rules as the operator’s staff.

Analysts of JRC-Karlsruhe are present on-site, ensuring a continuous flow of samples and results. The laboratories receive samples from all the plants on the respective sites, with dissolved spent fuel, plutonium products – including mixed oxide fuel – and inventory samples being the most important types of material from an accountancy viewpoint. Since the start of the On-Site Laboratories almost 15000 samples were received. The obtained analytical data are used by DG ENER for direct comparison with the operator’s declarations and allow the evaluation of the material flows and balance in a timely manner.

## 2.1 Methodology

As the nominal output of Pu can reach the order of a few tonnes per year for the larger reprocessing plants, safeguards inspectors require high-accuracy measurements in order to provide assurance of the absence of diversion. Sample measurement methods were selected using the criteria of highest possible measurement accuracy [2] and a minimum of resource consumption. Most of the samples are analysed by radiometric methods: K-Edge Densitometry (KED) for plutonium or uranium concentration; X-Ray Fluorescence (XRF) for the uranium/plutonium ratio; and High Resolution Gamma Spectrometry (HGRS) for the analysis of individual uranium and plutonium isotopes. The techniques of KED and XRF may be employed in combination in what is called Hybrid K-edge (HKED).

In order to determine the U/Pu amount content in absolute terms, the radiometric X-ray techniques must be calibrated against a standard which is traceable to the SI unit. Isotope Dilution Mass Spectrometry (IDMS) was chosen as the primary technique, and also serves for quality control of the radiometric methods due to its superior accuracy. IDMS is more labour intensive and is therefore typically carried out only on a subset of about 10 % of the samples and allows measurement of both the uranium and plutonium

concentrations and the respective isotopic compositions. Thermal Ionisation Mass Spectrometry (TIMS) is used for analysis of individual uranium and plutonium isotopes.

Quality assurance measures – both internal and external quality control – are of particular importance for accountancy measurements at facilities with a large throughput of nuclear material. The work at the On-Site Laboratories is performed according to quality management principles and follows the requirements of an ISO 17025 accredited laboratory. Analytical methods and procedures are whenever possible being improved and the laboratories are benchmarked through regular participation to inter-laboratory comparison exercises, such as EQRAIN (organised by CETAMA, France) and REIMEP (organised by JRC-Geel, Belgium). The On-Site Laboratories aim to conform to the latest international standards and achieve these in daily operation.

## 2.2 Laboratory infrastructure and analytical capacity

The On-Site Laboratories were designed with sufficient capacity, both in terms of instruments and staff, to analyse a certain number of samples. This number relates to the reprocessing capacity of the plants, to nuclear material accountancy regulations, the safeguards verification requirements, the material parameters to be measured, and the capacity of the different analytical methods. The analytical facilities are operated for about 45 weeks a year by typically 2-4 analysts a week. During operation, a minimum of two analysts must be present in the On-Site Laboratories according to safety-at-work rules.

The type of samples the On-Site Laboratories would receive was also identified during the conception phase. The newer reprocessing plants, such as UP3 and UP2-800 at La Hague and THORP at Sellafield, have different sample-taking regimes than the older Magnox plant. The sample types foreseen to be analysed dictated the layout and infrastructure of the individual laboratories.

### 2.2.1 The OSL laboratory at Sellafield

The OSL Sellafield receives samples both from the front-end but mainly from the back-end of the reprocessing cycle. The radioactivity levels are such that samples can be manipulated in the laboratory’s glove boxes. The sample types are diverse and most of them require labour-intensive sample preparation steps. The OSL is located in a building which is part of Sellafield’s Analytical Services. It comprises two ‘active’ laboratories, one ‘cold’ laboratory and an office space. All work with radioactive substances is performed in the two ‘active’ laboratories in glove boxes. There are 10 glove boxes of which the key boxes are:

- The “non-destructive analysis” glove box, where initial processing, analysis and sample preparation operations are carried out on product materials. Nearly all samples are analysed in this box. The glove box contains an

analytical balance, a microwave oven (for dissolutions), a densitometer and the equipment to perform radiometric analytical measurements (HKED, XRF, HRGS).

- A glove box dedicated to preparation of reference materials for mass spectrometry and for quantitative dilution of samples. The box is equipped with a bagless transport system to transfer material to the chemical separation boxes that are also equipped with this system.
- Each laboratory houses a chain of glove boxes for mass spectrometry. The chain consists of a box used for chemical separation of uranium and plutonium from fission products and other actinides and for alpha spectrometry, and the glove boxes for preparing the mass spectrometer's sample holder. The mass spectrometer is connected to the sample preparation box so that the prepared aliquots can be introduced directly into the mass spectrometer.

Samples taken at the front-end of the reprocessing plants – at the Head End Accountancy Tanks –, so called 'Input samples', are spiked in the Sellafield Thorp High Active hot cells. Only fractions sufficiently diluted to radiation levels acceptable for processing in a glove box are sent over to the OSL for chemical separation and mass spectrometry.

### 2.2.2 The LSS laboratory at La Hague

The majority of the samples received in the LSS La Hague are highly active input liquor samples. The LSS is located in an annex building to the UP3 plant. It houses three active laboratories and an office.

- The "product laboratory" is equipped with a glove box for preparation of uranyl-nitrate samples for K-edge and mass spectrometry measurement, and to prepare mass spectrometry reference solutions. The laboratory further hosts a suite of five boxes partly equipped with master-slave manipulators, and is dedicated to the measurement of isotopic compositions of PuO<sub>2</sub> product samples. Samples are received directly into the box by pneumatic transfer. The suite is connected to a gamma detector, and houses a hotplate and balance for dissolution and dilution of samples prior to TIMS measurement.
- The "hot cell facilities": The very high beta-gamma activities of input solutions and some product samples require the use of well-shielded hot cells equipped with master-slave manipulators. Because of the large number of input samples of different origin, and to avoid cross contamination, the hot cell suite consists of three interconnected hot cells which are all equipped with HKED spectrometers, balances and density measurement devices. The hot cells are connected to the plant's pneumatic transfer for automated sample receipt and to a liquid waste tank for the disposal of measured sample material. A small glove box is available for storage and treatment of reference materials.
- The "IDMS laboratory" consists of a suite of four glove boxes. The chain consists of a box for reception of

diluted samples from all other LSS facilities by pneumatic transfer, a box used for chemical separation of uranium and plutonium and for alpha spectrometry, a box for the assembly of the mass spectrometer's sample holder, and a box that allows the introduction of the sample holder into the mass spectrometer.

## 3. Methodology changes in the early years

The Euratom On-Site Laboratories have been operated successfully since their start-up. Naturally, there were a number of unexpected difficulties to which the laboratories had to adapt. Whenever necessary, methods and procedures were changed in order to improve quality of results and/or overall efficiency.

### 3.1 Reference Materials

Analytical methods need certified reference materials (CRM) traceable to the international standard (the mole) for (their) calibration. The chemical method installed in the On-Site Laboratories, the isotope dilution (ID) method coupled with TIMS, needs the addition of a "spike" for quantitative analysis since the signal intensity in TIMS is not proportional to the sample amount. A spike is an exactly known amount of target element with a different isotopic enrichment in one or more isotopes.

The original plan was to prepare various suitable CRMs and spike solutions in the JRC-Geel, Belgium and JRC-Karlsruhe for shipment to the On-site laboratories. Due to long delays in the organisation and delivery of the radioactive materials and most importantly due to the instability of especially plutonium solutions this soon turned out to be highly inefficient. The On-Site Laboratories had to revert to varied alternatives such as preparation on-site from solid primary material standard and spike solutions. Again, long term stability of such solutions together with the increased workload led the On-Site Laboratories to exclusively use the JRC-Geel large-sized dried spike (IRMM 1027 series – LSD) for traceability to the mole both for the verification of reference solutions (prepared from actual sample material) and as spike for isotope dilution. The quality and reliability of these LSDs determine the accuracy to which the laboratories can operate.

### 3.2 Analytical method changes

#### 3.2.1 Separation chemistry in the OSL Sellafield

A fundamental step of the sample preparation for alpha spectrometry and mass spectrometry analyses is the separation of uranium and plutonium fractions from fission products and minor actinides in the various sample types. A fully automatic system, developed by JRC-Karlsruhe, based on a Zymark robot, was originally installed in both On-Site laboratories. The separation method implemented in the OSL Sellafield was based on the PUREX process,

a liquid-liquid extraction method. By the time the LSS La Hague was constructed, a robotised method using chromatographic separation on resin (UTEVA®: Eichrom) had been developed [5]. The main advantage of the chromatographic separation method is a better recovery of U and Pu, meaning a smaller quantity of sample is needed, and no ozone-depleting reagents are used.

When the lifetime of the robot installed in the OSL came to an end, the OSL abandoned the PUREX process in favour of the UTEVA® chromatography.

### 3.2.2 Additional pre-separation chemistry in the LSS La Hague

IDMS plays a key role in the On-Site Laboratories as part of the quality control system. Strict limitations apply to the radiation level of samples allowed to be treated in glove boxes. An improvement which highly contributed to the quality of the IDMS technique involved the pre-separation of fission products from the input sample in the LSS hot-cell enclosures. This allowed transfer of less-dilute solutions, containing higher quantities of Pu and U, allowing comfortable chemical separation of the two elements and simple mass spectrometry measurements on each fraction [6].

## 4. Refurbishment projects

After the On-Site Laboratories had been running without interruption for some 10 years, it became apparent that some refurbishments and renewals were due in order to keep the laboratories functional, and to guarantee that the instrumentation remains up to the latest standard allowing high quality measurements. A major project started in 2010 with the replacement of a thermal ionisation mass spectrometer in the OSL Sellafield. Currently, a modernisation of the hybrid K-edge measurement system is ongoing in both On-Site laboratories, and further upgrades of mass spectrometers are planned for 2017-2018.

Refurbishments in the On-Site Laboratories tend to be rather complex. As a result of the multilateral nature of the On-Site Laboratories projects (DG ENER, JRC-Karlsruhe, site operator, and in the case of refurbishments often also a contractor having to work on the site to install new equipment), clear agreements and contracts have to be established. DG ENER is the owner of the laboratories and provides the necessary budgets for operations, routine maintenance, and refurbishments. The On-Site Laboratories are working under the respective Site Licences of the site operator and have to meet the requirements defined in these licences. They furthermore must comply with site specific safety rules, and follow site and building procedures. The site operators have specific procedures to be followed before any non-routine work is allowed to take place, and permission is needed before any new equipment may be installed. As a result, a large amount of paperwork

is to be completed, for review and acceptance by the site operator. In most cases, the site operator is involved in the execution of the work, as they are responsible for the provision and operation of the infrastructure. It is challenging to set up a sound time schedule and align the activities of all the stakeholders. Minor refurbishments, such as the installation of a small piece of new equipment, or a like-for-like replacement, are usually handled directly between the On-Site Laboratories analysts and the relevant site operator's staff. For larger projects, it is necessary to set up a dedicated contract to engage a site operator's projects team.

### 4.1 IT Hardware replacement project

The first substantial refurbishment project done in the On-Site Laboratories was the replacement of the IT hardware. The On-Site Laboratories Laboratory Information Management System (LIMS) was developed in the 1990s in the JRC-Karlsruhe, and consisted of a combination of software modules running under the OS/2 WARP operating system. The original computer network was put in place during installation of the On-Site Laboratories and had been running for almost ten years. Modernisation had been blocked until the OS2 version of Windows XP became available, allowing an economic solution for the upgrade. The LIMS software packages were migrated from OS/2 WARP to a Web based Microsoft XP using emulation software (Virtual PC). All new hardware was prepared and tested in JRC-Karlsruhe and thereafter sent to the On-Site Laboratories for installation. In the OSL Sellafield, the computer renewals also required an upgrading of the computer cables by the site operator.

### 4.2 Hybrid K-edge related projects

Improvements, developments, upgrades and renewals of the Hybrid K-edge system have been a continuous process since the early years of the On-Site Laboratories. Improvements were done to increase the efficiency of the laboratory and inevitably some broken equipment had to be replaced. Measures were taken to put a fall-back option in place to keep the OSL Sellafield operational in case of a fatal failure of the "non-destructive analysis" box. In the last couple of years preparations started for a complete modernisation of the Hybrid K-edge systems.

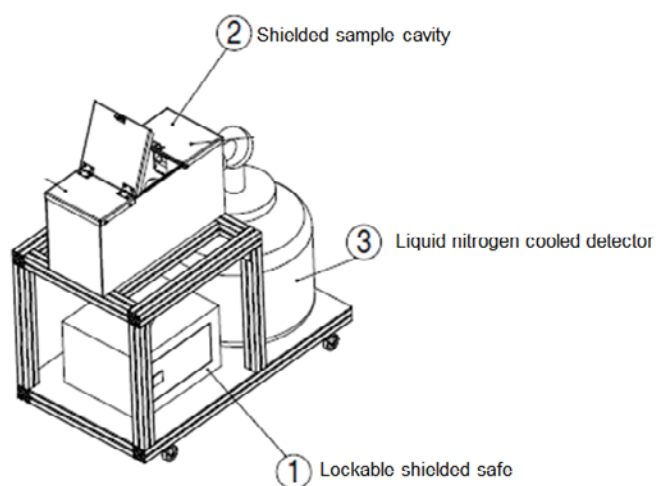
#### 4.2.1 Installation of an independent gamma station (OSL Sellafield)

Gamma spectrometry combined with Multiple Group Analysis (MGA) spectrum evaluation is a non-destructive means of measuring the  $^{238}\text{Pu}/^{239}\text{Pu}$ ,  $^{240}\text{Pu}/^{239}\text{Pu}$  and  $^{241}\text{Pu}/^{239}\text{Pu}$  isotopic ratios present in a sample and is the only practical means to quantify the americium/plutonium ratio via the  $^{241}\text{Am}/^{241}\text{Pu}$  nuclide ratio. Owing to the exceptionally low  $\gamma$ -activity of  $^{242}\text{Pu}$ , the  $^{242}\text{Pu}/^{239}\text{Pu}$  ratio cannot be measured, but the ratio can nonetheless be estimated using isotope correlation data [7, 8].



Sample throughput in the OSL Sellafield is limited by the "non-destructive analysis" glove box, where nearly all product samples are analysed. Originally, also the gamma measurements took place in this glove box. However, gamma measurements could only be carried out when the x-ray generator of the Hybrid K-edge/XRF was switched off. In addition, the  $\gamma$ -spectrometer arrangement in that glove box offered no means for optimisation of detector count rates, leading to exceptionally long counting times for MOX samples and to prohibitively long counting times for lower-activity solutions (e.g. oxalate mother solutions). For a gamma measurement there is actually no need to remove the sample from its protective packaging, and, provided that the samples are adequately double bagged, the samples can be measured outside the confinement of a glove box. To improve the sample throughput, to prevent an unnecessary shutdown of the X-ray generator and to allow all samples, including low-activity oxalate mother solutions, to be measured to better accuracy and with a reduced measurement time, it was decided to equip the OSL Sellafield with an external gamma station similar to the external gamma stations developed and in use at the JRC-Karlsruhe.

The external gamma station was manufactured in JRC-Karlsruhe. It consists of a high purity germanium detector, connected to a shielded sample cavity. The station is mounted on a trolley. Pending samples are stored in a lockable shielded safe. The external gamma station is connected to the signal processing electronics traditionally used for the glove box based gamma measurements. The gamma measurements, evaluation of the spectra and reporting of measurements to the LIMS are therefore under control of the Alpha Workstation.



**Figure 1.** Drawing of the independent gamma station.

The use of the external gamma station in the OSL Sellafield represents a considerable improvement for the non-destructive analysis routine work in terms of sample throughput and quality of measurement and in terms of work organisation and flexibility. The station enables the

whole sample to be viewed rather than a highly collimated portion of it; the gamma detector receives a noticeably higher amount of gamma rays, offers better count rate control and leads to quicker and more precise analysis. Also, the use of the external gamma station drastically improves the flexibility of the analysts work planning as gamma measurement can be performed in parallel to sample measurements involving K-edge/XRF.

#### 4.2.2 Refurbishment of two glove boxes (OSL Sellafield)

In the OSL Sellafield nearly all product samples are prepared in the "non-destructive analysis" glove box and measured by the equipment inside or connected to the glove box using radiometric techniques. The high radiation of the samples as well as the acids used for sample preparation has caused and may continue to cause degradation of the glove box. The glove box will continue to be used with improvements and equipment repairs as needed. However, an alternative analysis route, based on chemistry/mass spectrometry, has been installed as a fall-back option in case of a major disruption in the "non-destructive analysis" glove box. This fall-back option was the subject of an "OSL glove box refurbishment project" involving DG ENER, the plant operator and the OSL, and consisted of the refurbishment of two glove boxes:

- Conversion of a uranium glove box which was used previously for preparation and initial analysis of pure uranium samples. The glove box was re-categorised as a uranium/plutonium box for performing second-step dilutions on sample aliquots to a concentration which is suitable for separation chemistry, and for preparation of reference solutions needed for the calibration of the OSL instrumentation.
- Conversion of the existing uranium/plutonium dilution glove box into a "dissolution-spiking" box. The glove box was refurbished to take over initial processing and sample preparation operations on all product samples: weighing, dissolution, dilution and spiking. No radiometric techniques are foreseen to be performed in this glove box.

All design was made in JRC-Karlsruhe and submitted to the plant operator for approval. Manufacturing of the new equipment and materials needed was mostly done in the JRC-Karlsruhe workshops, and thereafter sent over to the OSL Sellafield for installation by the OSL analysts. One of the most challenging tasks was to design all materials such that they could be introduced in the glove box via the existing posting ports, e.g. a new glove box floor. Some of the innovative design is described further in this paper.

#### 4.2.3 Modernisation of the Hybrid K-edge densitometry system

The Hybrid K-edge / X-ray Fluorescence Densitometer instrumentation is used by Euratom and by the IAEA for Nuclear Material Accountancy Measurements [9]. It is the

instrument of choice for Safeguards measurements at Nuclear Fuel Reprocessing Plants such as Sellafield, La Hague and Rokkasho. The instrument was developed at JRC-Karlsruhe, some 30 years ago and continues to play an important role in Nuclear Safeguards measurements.

Although most of the instrument's original OpenVMS-based software is still very capable, the instrument's hardware is getting very old. The Alpha workstation computers, which run the software, and the spectroscopy electronic modules, which measure the K-edge and XRF spectra, are no longer made and must be replaced by modern equivalents. However, a ready-to-use combination of all modules replacing the whole interface cannot be purchased on the market. An innovative solution needed to be worked out in JRC-Karlsruhe. A further improvement of hardware tackled at the same time is the replacement of the traditional liquid nitrogen-cooled HRGS detectors by electrically cooled detectors.

#### 4.2.3.1. Installation of emulated Open VMS software

The Virtual Alpha emulator is a software application installed on a PC with Microsoft Windows® 7 OS (Host system), which emulates the functions of an Alpha workstation with OpenVMS operating system (Guest system). The guest system is saved as a virtual hard drive. This virtual hard drive is an image of a real hard drive of the VMS system running on an Alpha workstation. The emulator software enables the Open VMS and all the VMS software installed in the virtual hard drive to run on modern and most advanced PC architecture (x86 and the latest Intel Processors) with Windows® OS although designed for the Alpha workstation architecture. Hence this allows continued use of VMS specific software (i.e. Canberra k-edge software, Neutron Counting software etc.) on modern hardware.

The system was first set up and tested in JRC-Karlsruhe in collaboration with an external contractor (Migration Specialties Europe, Tarthorst, Netherlands). Meanwhile most of the Alpha workstations in the On-Site Laboratories have been replaced.

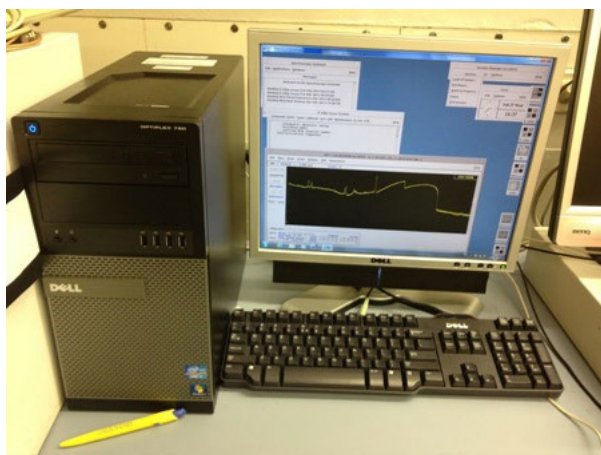


Figure 2. Virtual Alpha emulator installed on PC.

#### 4.2.3.2. Replacement of NIM electronic modules by Lynx

The traditional Nuclear Instrumentation Modules (NIM) are no longer manufactured and the repair of broken NIM units is no longer guaranteed. The Lynx hardware produced by Canberra is a good alternative to traditional NIM. A software driver to allow Lynx to run with the Hybrid K-edge/XRF's OpenVMS operating system has been developed and tested in JRC-Karlsruhe, so the Hybrid K-edge/XRF software is now capable of using modern Lynx spectroscopy hardware. The Lynx hardware outperforms the traditional NIM hardware except for digital peak stabilization. The HKED system uses two peaks (22.1 and 88.04 keV), originating from a  $^{109}\text{Cd}$ -source located close to the detector, as reference peaks for the digital stabilisation of the electronics. Properly peak-stabilized spectra are a crucial requirement for reliable Hybrid K-edge/XRF spectrum evaluation, and it is essential that this problem is remedied before the Lynx hardware can be used to replace the aging NIM electronics. An algorithm to improve upon the deficient Lynx peak stabilization has been developed at JRC-Karlsruhe. The computational approach performs better than the Lynx peak stabilization and even performs better than NIM-based peak stabilization. Tests on the post-processing algorithm have been completed, and the algorithm now has to be integrated into the OpenVMS-based Hybrid K-edge/XRF software. An additional advantage is that the new algorithm will be able to cope better with weaker  $^{109}\text{Cd}$  radio-active sources.

#### 4.2.3.3. Replacement of the X-ray generators (LSS La Hague)

The LSS has 4 Hybrid K-edge densitometry systems each equipped with a high voltage generator of 160 kV and 15 mA and X-ray tube control unit. The generators have exhibited increased breakdowns and reduced reliability. It is furthermore becoming more and more difficult to repair the instruments as spare parts are lacking. Merion – Canberra have indicated that future repairs cannot be guaranteed. New HV generators (GE Titan Isovolt) capable to deliver 160 kV and up to 15 mA and compatible with the existing X-ray tubes (Comet MIR 160/12) were ordered and will be installed in the near future. The replacement will ensure reliable operation for many years to come.

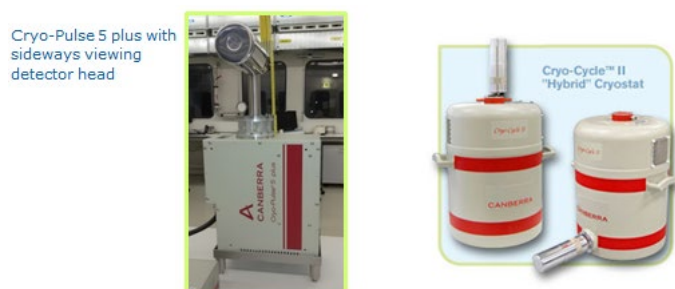
#### 4.2.3.4. Installation of electrically-cooled detectors

Replacement of the traditional liquid nitrogen-cooled detectors by electrically-cooled detectors eliminates the need for liquid nitrogen, which has the following advantages:

- Independence from the availability of liquid nitrogen in the controlled areas of the operator site
- Elimination of the risks of working with liquid nitrogen (frostbite, asphyxiation)
- Use of an environment-friendly alternative to liquid nitrogen
- Significant savings in running costs (labour, liquid nitrogen)

In the OSL Sellafield three detectors with their associated 25L Dewars will be replaced by three Cryo-pulse® CP-5 U-style Ge detectors. The Canberra Cryo-Pulse® 5 Plus is an electrically-powered cryostat for use with HPGe radiation detectors. It utilizes a pulse tube cooler, a highly reliable technology originally used in military and space applications, which has proven its value for germanium detectors in the original Cryo-Pulse 5. As stated by the producer, like its predecessor, the Cryo-Pulse 5 Plus consists of a cold-head-assembly, to which the detector is attached, and an external power controller. The basic external design and interface of the cold-head have been preserved to maximize interchangeability between the previous and the new version. However, the cold-head internals and the controller have been completely redesigned and new features have been added to improve the performance and reliability and to better answer customers' requirements. In order to deal with the restricted space under the "non-destructive analysis" glove box and in the external gamma station, the CP5 have been custom built to offer a sideways-viewing detector head. The cryostats will be placed on metal support frames, manufactured in the JRC-Karlsruhe, so that the units can be positioned in the exact required location. The cryostats have been tested in the JRC-Karlsruhe prior to shipment to the OSL. They are ready for installation as soon as the administrative procedure to obtain permission from the site operator will be finalised.

A similar replacement action is foreseen for the seven HPGe detectors at the LSS La Hague because two of the detectors cannot be repaired anymore. As space restrictions are not an issue, the LSS opted for the hybrid detector cooling system, a combination of an electrically-cooled detector with a liquid nitrogen cryostat, the Canberra Cryo-cycle II. Cooling is guaranteed by the liquid nitrogen in case of power outages. The cryostat can do without extra nitrogen for at least 6 months. A substantial reduction of liquid nitrogen consumption is to be expected.



**Figure 3.** Cryo-pulse® CP-5 U-style Ge detectors (left) for the OSL and Cryo-cycle II Ge detectors (right) for the LSS.

#### 4.2.4 Replacement of K-edge cooling

In both On-Site Laboratories, the refrigeration chillers which serve to cool down the X-ray tubes have been replaced when they became unrepairable. In the LSS La

Hague, a preliminary infrastructure study was performed by the JRC-Karlsruhe, the work itself was outsourced to a contractor. In the OSL Sellafield, the exchange of the refrigeration chiller was handled by the site operator's project team.

#### 4.2.5 Replacement of Anton Paar densitometers

The densitometry instrumentation currently in use is the Anton Paar DMA 48 in conjunction with an external measuring cell DMA 401. Both the DMA 48 and the DMA 401 are equipped with a borosilicate glass U-tube. The resolution of the instrument is  $0.00001 \text{ g/cm}^3$  and the accuracy can be as good as  $0.00005 \text{ g/cm}^3$ . The internal measuring tube is not used. The external measuring cell requires some modification before installation in the hot cells in the LSS and a glove box in the OSL. To maintain the temperature at  $25 \text{ }^\circ\text{C}$  (in the LSS) or  $20 \text{ }^\circ\text{C}$  (in the OSL) a temperature control circuit is used.

The DMA 48 and the DMA 401 are no longer available on the market. JRC-Karlsruhe has successfully used the spare internal measuring cells to manufacture external measuring cells. Now, there are no cells available anymore. The densitometers currently available on the market are integrated within a desktop model. A modification of these models for use in a nuclear environment is not possible since too many important components are not resistant to radiation. Anton Paar offers a new external measuring cell with a hollow U-tube made of Hastelloy C-276 (DMA HPM) to be used in conjunction with the mPDS5 evaluation unit. JRC-Karlsruhe is modifying and testing this instrument for future use in the On-Site Laboratories.

### 4.3 Mass spectrometry refurbishment projects

Mass spectrometry serves a twofold purpose at the On-Site Laboratories. Thermal Ionisation Mass Spectrometry is used to measure the uranium and plutonium isotopic compositions. Isotope Dilution Mass Spectrometry is employed to determine uranium and plutonium mass fractions using a well-characterized reference material, for example, "Large-Sized Dried Spikes" (LSD) [10, 11]. The latter results are also used to characterise the calibration solutions for the HKED densitometers.

#### 4.3.1 Decommissioning of an old TIMS MAT261 mass spectrometer and installation of a Triton

In 2009, a project was set up to replace a broken down MAT261 mass spectrometer in the OSL Sellafield, and replace it with a Triton (Thermo Fisher Scientific GmbH) instrument. It was the first major refurbishment project in the OSL, involving four parties (DG ENER, JRC-Karlsruhe, Sellafield Ltd. and Thermo Fischer). A dedicated contract was signed between DG ENER and the site operator for their Projects' team to bring the project forward and perform the necessary infrastructure works. The purchase of



the Triton was handled by JRC-Karlsruhe on behalf of DG ENER. Additional equipment such as a dedicated glove box was developed and manufactured in the JRC-Karlsruhe.

The following on-site work phases were identified:

- Disconnection of two glove boxes from the mass spectrometry chain and their decommissioning
- Decommissioning of the broken MAT 261 mass spectrometer
- Preparation of the laboratory infrastructure to accommodate the Triton specifications
- Installation of the new mass spectrometer
- Installation of a new glove box and connection to the separation chemistry box and the new mass spectrometer
- Commissioning and validation

With the installation of the Triton the OSL acquired a state-of-the-art instrument for mass spectrometry. The number of samples that can be loaded on one magazine is significantly higher than for the MAT261, and the instrument can run in independent mode, allowing overnight measurements. Hence, due to the Triton the OSL managed to increase the efficiency of the laboratory with delivery of high quality results.



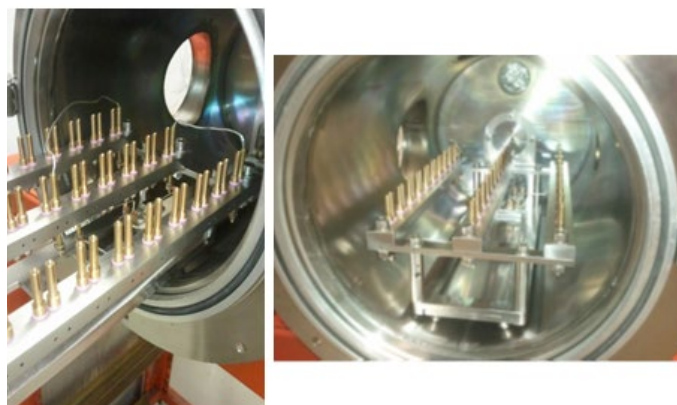
**Figure 4.** Triton connected to glove box (left) installed in the OSL and photo of the inauguration (right).

#### 4.3.2 Upgrade of the Filament Degassing Unit (OSL Sellafeld)

The Filament Degassing Unit (FDD) is needed to degas filaments before they can be used as sample holders for mass spectrometry. The FDD installed in the OSL Sellafeld was a 30 years old instrument with limited capacity. Shortly after the installation of the Triton, the Filament Degassing Unit was refurbished:

- Upgrade of the electronics
- Replacement of the filament rack, the new rack containing 30 positions

Since the upgrade, the Filament Degassing Unit is more reliable, with the main advantage that 30 filaments can be baked at once, saving valuable time for the mass spectrometry analyst.



**Figure 5.** Filament rack containing 30 positions.

The FDD unit in the LSS La Hague has become faulty and can only work on one side, thus degassing only half of its capacity. It will be replaced by a new instrument in the near future.

#### 4.3.3 Upgrade of three TIMS MAT26x mass spectrometers

The JRC has three TIMS MAT261 mass spectrometers operational: one in the JRC-Karlsruhe; one in the LSS La Hague; and one in the OSL Sellafeld. Due to their different ages (from 18 to 35 years old), the hardware/software components and the operation of the instruments is somewhat different. For the oldest instrument, located in the JRC-Karlsruhe, electronic modules are becoming sparse and it has become difficult to purchase original spare parts. Despite its age, the instrument remains important for training new analysts before they are allowed to work in the On Site Laboratories. Therefore it was decided to upgrade the three mass spectrometers. The company Spectromat provides commercial options for refurbishment of MAT26x instruments, such as the provision of modern electronics for operation and data acquisition, and some hardware components. The installation of Spectromat software is required for communication with the new electronic modules. Due to the different ages of the three mass spectrometers, the upgrades will be individually different: from new software only for the OSL instrument, to some hardware plus software upgrade for the LSS instrument; and of a complete exchange of electronics, some hardware and software in JRC-Karlsruhe.

The approach will allow the three instruments to remain in operation over the coming 10+ years, at a cost which is far lower than the purchase cost of new instruments. Moreover, all three mass spectrometers will be running with exactly the same software, which simplifies the work of the analysts.

## 5. Innovating developments

Over the years, the On-Site Laboratories faced specific problems which required creative solutions. Also in the framework of the extended refurbishment projects, some

ideas were worked out and led to many innovative developments that could be relevant for other laboratories.

### 5.1 Modifications to the densitometers located in hot cells (LSS La Hague)

To introduce a liquid in the glass U-tube of the external measuring cell of the Anton Paar density meter the cell is equipped with a screwed-on metal tube with an external diameter of 1.6 mm. It is however extremely difficult to mount a new tube or new flexible tubing. Although very resistant, the flexible tubing occasionally needs to be replaced. Also, the Teflon tips connected to the metal tube making contact with the glass cell break off after a certain time. The output side of the glass cell is connected to a 3-way valve via a similar system. A breakdown of this tube connection system usually meant a long downtime for the density meter as only a very experienced analyst was able to perform the repair. Since no useable alternative was available on the market, JRC-Karlsruhe has developed a new flexible tubing connection system for the glass cell, and quick-snap connections in stainless steel for the 3-way valve.



**Figure 6.** Flexible tubing connection for use with manipulators (left) and small stainless steel quick snap connections (right).

### 5.2 Infrared heater for preparation of alpha planchets

Alpha spectrometry measurements are required for all Pu fractions to be measured by mass spectrometry in order to de-convolute any possible isobaric interference from  $^{238}\text{U}$  with  $^{238}\text{Pu}$ . The system originally installed in the On-Site Laboratories for the preparation of alpha measurement planchets (metal sample holders) was prone to frequent breakdowns of its heating element. The system was also limited to a maximum planchet temperature of 180 °C, and planchets had to be prepared one at a time. JRC-Karlsruhe developed and manufactured a stand-alone heating unit containing a 100 W infra-red element. The unit includes safety features to prevent the inadvertent contact of the heating platform with glove box gloves or other combustible material. It also incorporates a timer which locks the unit until sufficient cooling time has elapsed before the analyst can access the heating platform. The new

unit delivers a planchet temperature of approximately 290 °C, which in turn prepares better quality planchets. As a result, almost no alpha planchets need repetition. Moreover, it is possible to prepare 4 planchets simultaneously, therefore reducing the overall preparation time.



**Figure 7.** Infrared heater for preparation of alpha planchets.

### 5.3 Semi-automated separation unit

A fully automated Zymark robot was originally installed in the On-Site Laboratories for chemical processing of the samples. Chemical separation is required to remove decay and fission products and provide separated U and Pu fractions for mass spectrometry. After ten years of use, the robots came to the end of their lifetime and repeated breakdowns led to long downtimes, while it became increasingly difficult to obtain spare parts after the commercially available device was withdrawn from the market. While routine operations continued with manual separations, a new solution was needed to replace the outdated robots. The device had to be safe to operate in a glove box, and be compact enough to allow posting it into the existing boxes via the posting ports. A semi-automated separation device for chromatographic separation on resin (UTEVA®: Eichrom) has been developed by the JRC-Karlsruhe in collaboration with the IAEA under the framework of the EC support programme to the IAEA [12]. The JRC-Karlsruhe planned and built the unit while the IAEA developed the controlling software based on an early version from JRC-Karlsruhe. The main features of the semi-automated separation unit are its modular construction for simple replacement of components; minimum need for operator intervention; its light structure, built using materials resistant to acid environment; and its remote control function via a LabView-based software. The benefits of the semi-automated separation unit are a reduction of the radiation dose rates in the vicinity of the operators and an increase in the sample throughput. The device is expected to be installed in the LSS La Hague in the near future.



Figure 8. Semi-automated separation unit.

#### 5.4 Foil heater, a hotplate based on a flexible low power heating element

A hotplate, for the purpose of dissolving spiked samples or powders/pellets in a glove box, has been designed, manufactured and tested at JRC-Karlsruhe. Heat is provided with a flexible heating element that can be purchased from several manufacturers (Minco, Termya, and Synomas). The element is made of wires insulated in a polyimide film (Kapton®). The operational voltage range employed to dissolve samples is between 16 V and 20 V: within this operational range the power dissipated per unit of the hotplate surface varies from 0.26 W/cm<sup>2</sup> to 0.32 W/cm<sup>2</sup>. Even lower voltages are suitable for spike-dissolutions. The heating element is enclosed in an outer casing made of polycarbonate with UL94 V-0 flammability certification. The external temperature on the surface of the outer casing reaches no more than 60 °C. Heat output and external-surface temperature of the foil heater enclosure are therefore acceptable for use inside a glove box. An outlet in the casing is connected to the ventilation system to remove acid fumes. The hotplate has inserts allowing secure placement of spike vials or various sizes of Erlenmeyer flasks. Several heating devices can be connected in series. The case(s) containing the flexible heating system and the temperature sensor are located in the glove box, whilst the power supply and the temperature control unit are located outside the glove box. The cabling is connected via feed-through "LEMO" connectors.

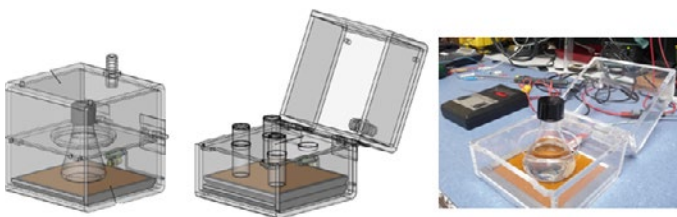


Figure 9. Foil heater for use with Erlenmeyer (left) or spike vials (middle) and prototype (right).

#### 5.5 Mounting of additional lead shielding to glove boxes

The glove boxes installed in the OSL Sellafield are Perspex type boxes. The glove box window panels and ceiling consist of 12 mm Perspex, and the bottom plate is made of stainless steel. The glove boxes are supported on a steel frame. During installation, those gloveboxes foreseen at the time to be used for more active samples were provided with additional 2cm thick lead glass window panels. In the framework of the OSL refurbishment, additional shielding was installed to some gloveboxes to provide increased protection to operators from radiation.

##### 5.5.1 Additional shielding surrounding an existing Perspex type glove box

The steel framework, on which the Perspex glove box is resting, had no features foreseen to add heavy additional shielding. A design was made to clamp a second support frame to the existing glove box support frame. The arrangement did not require welding or drilling. The shielding was comprised of half-thickness leaded-panels (in front of the Perspex side-windows), with painted lead shielding fitted to the fourth side where a bagless transport system was in place.

##### 5.5.2 Additional shielding underneath an existing glove box

Additional lead shielding was installed underneath two OSL Sellafield glove boxes. Lead sheets with a thickness of 2-3 mm (supported by a 2 mm layer of aluminium) were inserted under the glove box, thereby increasing the shielding efficiency of the glove box's floor and protecting the lower half of the operator's body from radiation uptake. The installation of the extra shielding required neither particular intervention nor modification of the infrastructure. The lead sheets slide in between the existing glove box frame and existing lead window frame so that they are positioned under the glove box but on the top of the glove box stand. The lead sheets are secured in place with steel clips and an additional support bar. The overall weight, estimated at 55 kg, is well within the tolerance limit of the glove box stand.

#### 5.6 Replacement of a glove box's flooring

In the framework of the refurbishment of a glove box in the OSL Sellafield, it was decided to replace the glove box flooring. Several pieces of equipment were removed during the cleanout of the box, leaving the original plastic floor uneven. Not only would this have posed some restrictions in the possible layout of the refurbished box, but also the risk of accidental spillages would have been considerable.

##### 5.6.1 Removal of the old flooring

The plastic floor sheet which covered 75% of the glove box floor had to be removed to allow the fitting of a new floor. JRC-Karlsruhe selected and trialed a powered cutting tool (Multimaster) for the cutting-up of the plastic floor sheet. Also,



some silicon sealant had to be removed whilst taking care not to damage any of the materials that form part of the glove box containment, such as the Perspex panels or the rubber sealant between the Perspex and metal frame. An aluminium adjustable spacer was designed and attached to the cutting tool, which controlled the depth of the cut with high precision. Tests were performed and videoed in the JRC-Karlsruhe workshop and submitted to the plant operator to prove the safety of the tool. Training was given to the analysts in the use of the Multimaster in a mock-up glove box in JRC-Karlsruhe prior to performing the work on site.

#### 5.6.2 Installation of new polycarbonate glove box flooring

The design of the new glove box floor was quite challenging:

- All materials needed had to be introduced into the glove box via the existing posting ports. Glove boxes in the OSL Sellafield have 8 inch ports, with one 16 inch port in the roof of the box.
- Only non-combustible or fire retardant and heat resistant materials are allowed to be used throughout the facility. The plant operator required the use of a UL94 V-0 certified material, which is difficult to obtain in Europe in small quantities.
- The design must allow regular inspection of the area underneath the floor for spillages, as well as the possibility to clean up such spills

The new glove box floor is made from polycarbonate rods that slot together to form a frame, then levelled and covered by polycarbonate tiles. The polycarbonate used is Makrolon GP Clear 099 with the required UL94 V-0 certification. Once in place, the middle tiles are removable with the aid of a rubber plunger. A trial assembly was carried out in the JRC-Karlsruhe workshop prior to the flooring being delivered to site.

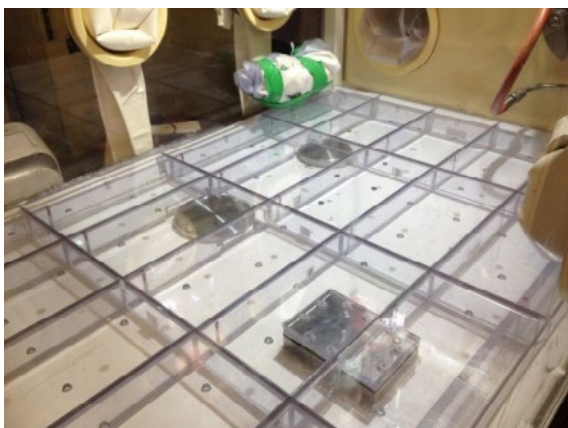


Figure 10. New polycarbonate glove box floor.

## 5.7 Replacement of feed-through LEMO connectors

The feed-through LEMO connectors originally fitted to the glove box panels in the On-Site Laboratories were

showing signs of corrosion and were going green in colour due to the material of manufacture and the glove box environment. JRC-Karlsruhe has developed two ways of addressing the problem.

#### 5.7.1 Installation of additional LEMO connectors via a glove or posting port

In JRC-Karlsruhe, additional LEMO feed-through connectors are installed by sacrificing a glove or posting port. An unused port is closed off by a port cap containing the necessary feed-through connectors. The system is also used for throughput of reagents tubing.



Figure 11. Posting port with LEMO feed-through connectors.

In the LSS La Hague, additional feed-through connectors will also be installed via a posting port. However, the choice fell on titanium connectors from Souriau (formally Jupiter) in line with the plant operator practices.

#### 5.7.2 Exchange of feed-through LEMO connectors (OSL)

In the OSL Sellafield, feed-through LEMO connectors were replaced during the refurbishment of two glove boxes. LEMO connectors which are no longer needed have been covered by stainless steel caps. All other LEMO connectors were replaced by stainless steel bulkhead LEMO plug/socket connectors. In the first glove box, the contamination levels were so low that, apart from wearing gloves and respirator protection, no additional measures were required. The contamination level in the second glove box called for a more stringent method to be developed. A new set of tools was specially developed, along with a specific methodology, the application of which would limit the need for cleaning to "contamination-free" levels to the area

immediately surrounding the feed-through connector. Within a procedure that is the subject of a current patent application, the arrangement provided engineered safeguards to guarantee that no contamination was released from the glovebox during the exchange of connectors.

## 6. Conclusions

The Euratom On-Site Laboratories have been operated successfully for more than 15 years. Based on the operational experience, processes have continually been optimised and procedures streamlined. After the first ten years, optimisation alone was no longer sufficient to keep the laboratories operational in the long term. Therefore, laboratory refurbishment had to be looked at with an open mind for future needs. Renewals in the On-Site Laboratories mostly involve specialised and fit-for-purpose equipment that cannot be purchased on the market without adaptations, or cannot be purchased at all. The analysts working in the On-Site Laboratories have always been dedicated to come up with creative, purpose-built, problem-oriented solutions. The JRC-Karlsruhe in-house design team and workshop have proved to be of utmost importance to support the On-Site Laboratories. This cooperation resulted in a whole series of innovative developments, which may be useful for other laboratories in the nuclear and/or other fields. Also, the site operators play an important role in any refurbishment project. The refurbishment programs are still ongoing, and are foreseen to continue over the coming years. The On-Site Laboratories are well placed to continue to deliver high quality results to DG ENER.

## 7. Acknowledgements

The authors would like to acknowledge the JRC-Karlsruhe design office and workshops for their invaluable contribution to all the innovations and refurbishments described. The authors also express their gratitude to the site operators of La Hague and Sellafield for their support.

## 8. Legal matters

### 8.1 Privacy regulations and protection of personal data

The author agrees that ESARDA may print her name/contact data/photograph/article in the ESARDA Bulletin/Symposium proceedings or any other ESARDA publications and when necessary for any other purposes connected with ESARDA activities.

### 8.2 Copyright

The author agrees that submission of an article automatically authorises ESARDA to publish the work/article in whole or in part in all ESARDA publications – the bulletin, meeting proceedings, and on the website.

The author declares that their work/article is original and not a violation or infringement of any existing copyright.

## 9. References

- [1] IAEA Report of the LASCAR Forum; *Large Scale Reprocessing Plant Safeguards*; STI/PUB/922, Vienna, Austria, 1992.
- [2] K. Casteleyn, L. Duinslaeger, P. Amador Celdran, P. Arboré, P. Chare, E. Dahms, O. Drinnhausen, H. Eberle, T. Enright, A. Guiot, M. Hild, J. Horta Domenech, A. Le Terrier, K. Lützenkirchen, S. Millet, S. Morel, M. Muñoz Nieto Sandoval, A. Nicholl, S. Nucifora, H. Ottmar, D. Poublan, G. Rasmussen, F. Sarli, J. Schönfeld, H. Schorlé, P. Schwalbach, S. Street, P. van Belle, E. Zuleger; *The Euratom Safeguards On-Site Laboratories at the Reprocessing Plants of La Hague and Sellafield*; JRC Reference Report; 2011; <https://ec.europa.eu/jrc/en/publication/reference-reports/euratom-safeguards-site-laboratories-reprocessing-plants-la-hague-and-sellafield>.
- [3] U. Blohm-Hieber, F. MacLean, P. Chare, W. Kloeckner, K. Mayer, H.G. Schneider, and H. Ottmar; "OSL" – *Inauguration of the first EURATOM On-Site Laboratory*; Proceedings of the 22nd ESARDA Annual Meeting, Dresden, Germany; 2000.
- [4] U. Blohm-Hieber, P. Chare, W. Kloeckner, P. Daures, H. Ottmar, P. Richir, G. Décobert, and X. Rincel; "LSS" – *Inauguration of the EURATOM "Laboratoire-sur-site"*; Proceedings of the 23rd ESARDA Annual Meeting, Bruges, Belgium; 2001.
- [5] P. Richir, B. Brandalise, C. Apostolidis, R. Molinet, and K. Mayer; *Development of a robotized separation method for U-Pu samples using UTEVA resin*; Proceedings of the International Workshop on the Application of Extraction Chromatography in Radionuclide Measurement, Geel, Belgium; 1998.
- [6] P. Richir, A. Guiot, A. Le Terrier, J. Horta, P. van Belle, and P. Daures; *Simple, robust and rapid method to separate the U-Pu fraction from an input solution in a hotcell facility prior to its transfer into a glove-box*; Proceedings of the 25th ESARDA Annual Meeting, Stockholm, Sweden; 2003.
- [7] R. Gunninck. W.D. Ruther; *MGA: A Gamma-ray Spectrum Analysis Code for Determining Plutonium Isotopic Abundances, Vol.1 (Methods and Algorithms)*; Report UCRL-LR-103220, Vol.1, April 3; 1990.
- [8] R. Gunninck. W.D. Ruther; *MGA: A Gamma-ray Spectrum Analysis Code for Determining Plutonium Isotopic Abundances, Vol.2 (A Guide to Using MGA)*; Report UCRL-LR-103220, Vol.2, September 1; 1990.

- [9] K. J. Moody, D. A. Shaughnessy, K. Casteleyn, H. Ottmar, K. Lützenkirchen, M. Wallenius, T. Wiss; *Analytical Chemistry of Plutonium*. In *The Chemistry of the Actinide and Transactinide Elements*, L. R. Morss, N. M. Edelstein, J. Fuger, Eds.; Springer: Dordrecht, Vol. 6, pp 3889-4004; 2010.
- [10] J. Berger, N. Doubek, G. Jammet, D. Donohue, G. Bagliano, K. Mayer, A. Verbruggen, C. Ingelbrecht, F. Hendricks, A. Alonso, P. De Bièvre, and H. J. Ahrenz; *Development and Experience with  $^{235}\text{U}$  and  $^{239}\text{Pu}$  Solid Spikes for IDMS of Spent Fuel Solutions*; Proceedings of the 17th ESARDA Annual Meeting, Aachen, Germany; 1995.
- [11] E. Zuleger, K. Casteleyn, L. Duinslaeger, K. Mayer, P. Richir, and P. van Belle; *Accuracy and Precision achieved in Thermal Ionisation Mass Spectrometry of Uranium and Plutonium in a Safeguards Analytical Laboratory under Routine Conditions*; Proceedings of the 45th INMM Annual Meeting, Orlando, USA; 2004.
- [12] R. Buda, S. De Vry Balsley, L. Emblico, A. Schachinger, E. Zuleger; *Semi-Automatic Separation Unit for Actinides at JRC-ITU and IAEA*; Proceedings of the 37th ESARDA Annual Meeting, Manchester, UK; 2015.







