ISOE International Symposium

Tours, 21 – 23 June 2022

Programme & Book of Abstracts

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ISOE International Symposium

Tours, 21-23 June 2022

The European Technical Centre of the Information System on Occupational Exposure (ISOE) is pleased to organize, in collaboration with and the support of Electricité de France (EDF), the 2022 ISOE International Symposium on Occupational Exposure Management at Nuclear Facilities.

The Symposium will be held in Tours, France, from the 21st to the 23rd of June 2022. It is co-sponsored by the OECD Nuclear Energy Agency (NEA) and the International Atomic Energy Agency (IAEA).

The Symposium is targeted at all those concerned with radiological protection at nuclear power plants: radiation protection managers and staff members, maintenance and operation planners, contractors, exposed workers, regulatory body representatives and international organisations. It is also opened to research reactors and professionals from other nuclear fuel cycle installations sharing common radiological protection issues.

Dealing with occupational radiation protection at the design, operation and decommissioning stages of installations, as well as accident situations, this new meeting point of radiation protection professionals under the heading of ISOE will be a great opportunity to share, at the international level, experiences and practices favouring a continuous improvement of radiation protection.

Furthermore, the accompanying Technical Exhibition will give participants the opportunity to see the latest developments from industrial and commercial companies active in fields of radiation protection.

Prior to the Symposium, on Monday 20 June 2022, two meetings devoted to specific audiences have been organised:

- A Radiation Protection Managers meeting
- A Regulatory Body Representatives meeting

We are looking forward to welcoming you in Tours,

Caroline SCHIEBER
Head of ISOE-ETC
On behalf of the Programme Committee

Programme Committee Members

Lucie D'ASCENZO	ISOE ETC, CEPN – France
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Thorsten STAHL	GRS – Germany
Torgny SVEDBERG	Ringhals NPP – Sweden
Philippe WEICKERT	EDF UNIE-GPEX – France

Conference Language

The conference language will be English.

Symposium Venue

The Symposium will take place at:

Palais des Congrès de Tours
26 Boulevard Heurteloup
37000 Tours

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DEDICATED MEETINGS – MONDAY 20 JUNE 2022

The ISOE Symposium will be preceded by two dedicated meetings devoted to specific audience.

If you wish to participate to one of those meetings, please contact the relevant person as indicated below in order to register to one of these meetings.

The registration to these meetings is free of cost.

Radiation Protection Managers Meeting	
09:00 - 17:00	Location: Palais des Congrès de Tours *Contact-person:* **Philippe WEICKERT**, EDF – philippe.weickert@edf.fr

Regulatory Body Representatives Meeting	
09:00 - 17:00	Location: Palais des Congrès de Tours *Contact-person:* **Charlotte GUENAULT**, ASN – Charlotte.GUENAULT@asn.fr

ISOE International Symposium

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PROGRAMME

TUESDAY 21 JUNE 2022

08:30 - 09:15	*Registration*

09:15 - 10:00	Opening Ceremony (ETC, OECD NEA, IAEA, EDF, ASN)

Session 1. Chairpersons	RP at the Design Stage *V. Rees (ONR, UK), C. Schieber (ISOE ETC – CEPN, France)*
10:00 - 10:20	Radiation Protection optimization at Design stage: application to EPR2 M. Longeot (EDVANCE/EDF), S. Poirrier (EDVANCE/Framatome), France
10:20 – 10:40	Computation of dose equivalent rates during the loading of spent nuclear fuel in a transport cask using Monte Carlo simulation F. Hoareau (EDF R&D), V. Buforn, C. Gaubert (EDF DIPDE), F. Thibaud (EDF DCN), France

10:40 - 11:25	*Coffee-break, Visit of Exhibition, Posters*

Session 2. Chairpersons	New RP Technologies *P. Arends (ANVS, Netherlands), T. Stahl (GRS, Germany)*
11:25 - 11:45	NextGen RP: Applying Remote and Automated Technologies to Enhance and Optimize Nuclear Power Plant Radiation Protection Operations D. Perkins, K. Kim, P. Tran, D. Cool (EPRI), USA
11:45 - 12:05	ECHO: an innovative device for the measurement of beta contamination in high and fluctuating gamma environments L. Bardou (EDF UNIE-GPEX), T. Le Noblet, V. Maussire (CARMELEC), France

12:30 - 14:00	*Lunch Break*

Session 3. Chairpersons	**Source-Term Management** *B. Breznik (Krško NPP, Slovenia), G. Ranchoux (EDF DP2D, France)*
14:00 - 14:30	ALARA tools: Source Term Control at CNE Cernavoda C. Chitu, A. Nedelcu (Cernavoda NPP), Romania
14:30 - 15:00	Impacts of Ag-110*m* on Radiation Field Generation: Review of an innovative experiment E. Moleiro, A. Rocher (EDF UNIE-GPEX), T. Tribollet (Civaux NPP), France
15:00 - 15:30	Application of Inline Gamma and CZT D. Perkins, J. McElrath, M. Mura (EPRI), L. Whiteker (Monticello NPP), USA

15:30 - 16:15	*Coffee-break, Visit of Exhibition, Posters*

Session 4. Chairpersons	**ISOE & UNSCEAR** *J. Ma (IAEA – ISOE co-Secretariat), C. Schieber (ISOE ETC – CEPN, France)*
16:15 - 16:45	30 Years of ISOE Global ALARA Sharing: Past, Present & Future B. Boyer, ISOE Chair
16:45 - 17:15	Occupational exposure to ionizing radiation: UNSCEAR Global Survey and ISOE DATA F. Shannoun, V. Holahan, UNSCEAR

WEDNESDAY 22 JUNE 2022

Session 5. Chairpersons	Job Experiences (Part I) *J. Bonnefon (EDF DIPDE, France), L.-A. Beltrami (ISOE ETC – CEPN, France)*
09:00 - 09:20	ALARA in Repair of Reactor Vessel Flange Surface at Krško NPP B. Breznik (Krško NPP), Slovenia
09:20 - 09:40	Case Report of Incident with High Exposure of Radiation J. Isokivelä (Forsmark NPP), Sweden
09:40 - 10:00	Radiation protection measures during the steam generator heat transfer tubes cleaning at Dukovany NPP J. Novak, R. Svoboda (Dukovany NPP), Czech Republic

10:00 - 10:45	*Coffee-break, Visit of Exhibition, Posters*

Session 6. Chairpersons	Experiments and new R&D Developments *H. Meijer (Borssele NPP, The Netherlands), P. Weickert (EDF UNIE-GPEX, France)*
10:45 - 11:05	3D-CZT Gamma-Ray Spectrometers and Imaging Spectrometers for Source Term Characterization at Palisades Nuclear Plant M. Mayers (H3D, Inc.), D. Nestle (Palisades NPP), W. Kaye (H3D, Inc.), USA
11:05 - 11:25	ASTRE Development S. Poumerouly, C. Monier, A. Geay, M. Charwath, S. Tonnoir, V. Lombard (EDF R&D), France
11:25 - 11:45	Development and Operations of the Ultra-Compact Embeddable Gamma Camera Nanopix F. Carrel, V. Schoepff, Y. Moline, G. Amoyal, J.P. Poli, M. Gendreau, M. Morenas, M. Imbault, J.M. Bourbotte, R. Woo, C. Lynde, M. Michel (CEA LIST), R. Abou Khalil, Z. Mekhalfa (Orano DS), C. Helbert, L. Tondut (ORANO Etablissement de la Hague), France

11:45 - 12:15	Synthesis of RPM and RB meeting days

12:30 - 14:00	*Lunch Break*

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Session 7. Chairpersons	**RP Aspects of Post-Accident Situations** *C. Guenault (ASN, France), R. Svoboda, (Dukovany NPP, Czech Republic)*
14:00 - 14:30	Health Effects from the Chernobyl and the Fukushima Accidents E. Cléro (IRSN), France
14:30 - 15:00	Ten years at the Fukushima Daiichi Nuclear Power Station, then and now T. Suzuki (TEPCO D&D Engineering Company), Japan
15:00 - 15:30	The role of the radiological protection expert in stakeholder involvement in the recovery phase of post-nuclear accident situations: Some lessons from the Fukushima-Daiichi NPP Accident T. Schneider (CEPN), France

15:30 - 16:15	*Coffee-break, Visit of Exhibition, Posters*

Session 8. Chairpersons	**Job Experiences (Part II)** *S.-G. Jahn (ENSI, Switzerland), V. Riihiluoma (STUK, Finland)*
16:15 - 16:35	Corium stabilization: a challenging Radiation Protection project C. Peretti (EDF DIPDE), France
16:35 - 17:15	ALARA Experience with PWR Thermal Sleeve Replacement at Sizewell B G. Whyberd-Mills, R. Parlone, G. Renn (Sizewell B NPP), United Kingdom

19:30	**Symposium Dinner at City Hall** Address: 2 Boulevard Heurteloup 37000 Tours	

THURSDAY 23 JUNE 2022

Session 9. Chairpersons	RP and Decommissioning *E. Thoelen (ENGIE Electrabel, Belgium), I. Calavia (CSN, Spain)*
09:00 - 09:20	ISOE Working Group on Radiological Protection Aspects of Decommissioning Activities at Nuclear Power Plants (WGDECOM) – Outcomes and feedback G. Ranchoux (EDF DP2D), L.-A. Beltrami (ISOE-ETC – CEPN), France
09:20 - 09:40	Decommissioning project, management and planning aspects for the decommissioning of Research Reactor Ispra1 E. Amoroso, D. Manes, S. Ravera, K. Harralabos, P. Capoferro, E. Grossi (SOGIN), Italy
09:40 - 10:00	Radiological investigations in hard-to-access zones during D&D operations by remote OSL/FO dosimetry S. Magne, M. Horpin (CEA LIST - Laboratoire Capteurs Fibres Optiques), W. Husson (CEA LIST - Laboratoire Capteurs Architectures Electroniques), S. Dogny (ORANO), P.G. Allinei, O. Guéton, M. Ledieu (CEA DEN - DTN-SMTA-LMN), B. Leibovici (SDS Group), France
10:00 - 10:20	Development of a Cask for Interim Storage and Final Geological Disposal in Switzerland Approach, Challenges and Realization S. Pudollek (Nagra), E. Neukäter, R. Graf (Mühleberg NPP), Switzerland

10:20 - 11:05	*Coffee-break, Visit of Exhibition, Posters*

Session 10. Chairpersons	RP Indicators *J. Isokivelä (Forsmark NPP, Sweden), C. Schieber (ISOE ETC – CEPN, France)*
11:05 - 11:25	Improving occupational radiation exposure using ALARA tools: performance indicators C. Chitu, A. Nedelcu, L. Samson (Cernavoda NPP), Romania
11:25 - 11:45	Dose per RCA-hour; a useful radiological protection indicator? R. Parlone, G. Renn (Sizewell B NPP, United Kingdom), B. Joerger (Penly NPP, France)
11:45 - 12:05	RP Management Recovery Plan at EDF: Indicators and Tools P. Weickert (EDF UNIE-GPEX), France

12:05 - 12:20	Distinguished Papers and Closure of the Symposium

BOOK OF ABSTRACTS

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POSTER PRESENTATIONS

Session 1. Radiation Protection optimization at Design stage: application to EPR2

Matthieu Longeot[1] (matthieu.longeot@edvance.fr), *Sébastien Poirrier*[2] (sebastien.poirrier@edvance.fr)
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[2] EDVANCE/Framatome - 76-78 Avenue Tony Garnier ; 69007 Lyon ; France

Following the development of the EPR technology, EDF Group (EDF SA and Framatome) is designing the EPR2 model which optimizes costs and construction delays based on lessons learned from previous EPR units (in operation and in project), improving in particular its constructability. EPR2 aims at having GEN3 safety and radiation protection qualitative objectives and quantitative objectives equivalent to EPR.
In this frame, EDVANCE (*) is in charge of the design of EPR2 nuclear island. As for all EDF Group nuclear projects, radiation protection is taken into account at early design stage of the project similarly to what is done for safety.

After an introduction showing how EDVANCE integrates regulatory and operator requirements in the design, the presentation focuses particularly on the EPR2 collective dose optimization. It explains the structured approach applied with the following steps.
First, EDVANCE computes the average collective dose per type of outages (“refueling only”, “normal” and “in-service inspection” outages) for the most efficient units of EDF French fleet. Then, based on EPR2 timing of outages, EDVANCE uses these operating experience (OPEX) values to estimate what is called the “EPR2 Reference Dose”.
Second, taking into account benefits of EPR2 basic design (and later detailed design) changes compared to EDF French fleet and operator good practices, EDVANCE determines the “EPR2 initial dose assessment”. Finally, activities and categories of workers with the highest dose and "dose rate" are identified. The objective is to optimize these activities by lowering the following parameters: a) source term, b) dose rate; c) duration of exposed work. The envisaged optimization measures will be presented.
The final outcome of the study is the “optimised EPR2 dose assessment”. Even if the study will be updated at detailed design stage, EDVANCE is confident that the optimization will allow reaching a collective dose lower than the EPR2 initial objective of 0.35 Man.Sv/yr/unit averaged over ten years put forward at conceptual design.

The presentation will finally focus on how EDVANCE defines cleanliness/waste zoning requirements at design stage of EPR2 in particular to limit the contamination at its source. These requirements ensure that radiological cleanliness is at the best international operators level and that the amount of waste sent to radiological waste streams is reduced. Moreover, these cleanliness/waste zoning requirements contribute to the reduction of transfer of contamination outside of the installation. The paper will present in particular the “EVEREST operational approach” considered as an EPR2 objective at design stage. With this approach, worker enters "in blue" in the clean zones. At all installation interfaces (also called “contamination barriers”) between clean and contaminated areas, the worker finds overclothes and a device to check contamination of personnel and equipment at the exit. With these design measures, the worker has personal protective equipment adapted to the level of contamination risk.

(*) EDVANCE is an EDF and Framatome subsidiary in charge of the design and construction of the nuclear island of the new built projects in which EDF is involved.

Session 1. Computation of dose equivalent rates during the loading of spent nuclear fuel in a transport cask using Monte Carlo simulation

F. Hoareau [1], V. Buforn [2]**,** C. Gaubert [2], F. Thibaud [3]
[1] EDF R&D, [2] EDF DIPDE, [3] EDF DCN

Spent nuclear fuel (SNF) is shipped from EDF nuclear power plants (NPP) to the La Hague facility in dedicated transport casks. During the loading of these casks, EDF must ensure that dose equivalent rates (DER) satisfy admissible criteria:
- In the loading area (the fuel building of the NPP)
- In the adjacent rooms of the fuel building
- Outside the fuel building
- In the public area (at the limit of the NPP)

EDF will soon use new “TNG3L” ORANO NPS transportation casks for the transportation of SNF in 1300 MWe and 1450 MWe NPP. These TNG3L casks will replace the currently operated “MARK II” casks. During the specific phase of SNF loading into the shipping cask, MARK II casks are added with a radiation biological shield that lower the DER to an acceptable level for the employees and the public. The radiation biological shield used for the MARK II casks cannot be used for TNG3L casks because of inconsistent geometries. Therefore, a new radiation biological shield (with a lead thickness of 6 cm) has been designed for the TNG3L casks.

In order to evaluate the efficiency of this new radiation shield, the loading area and its surroundings (fuel assemblies, TNG3L cask, radiation shield, buildings and exterior) has been modelized with the Monte Carlo transport code TRIPOLI-4® (developed by CEA). Particle transport (neutron and gamma) can then be simulated in 3D to compute DER in the areas of interest.

The performed calculations (with a lead thickness of 6 cm for the shield) have shown that the criteria within the accessible parts of the loading area, the adjacent rooms and the public environment are satisfied. Subsequent calculations with reduced lead thickness have shown that a lead thickness of 4 cm for the shield is sufficient to satisfy the admissible DER criteria.

This study has thus led to an optimization of the lead thickness of the radiation shield. It will also help the NPP operators to set up the preliminary radiation zoning during the loading of TNG3L casks

Session 2. NextGen RP: Applying Remote and Automated Technologies to Enhance and Optimize Nuclear Power Plant Radiation Protection Operations

David Perkins[1], Karen S. Kim[1], Phung K. Tran[1*], and Donald A. Cool[2]

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ABSTRACT

Currently, most radiation measurement and characterization activities that occur at nuclear power plants are conducted manually and on a routine basis regardless of whether conditions warrant the evaluation. Advances in sensor, indoor position systems, and data transmission science and technology have enabled remote and automated operations in many industries. There have been advances in radiation remote monitoring technology (RMT) for non-nuclear power plant purposes (for example, for security purposes) and for environmental monitoring following the Fukushima accident. The combination of remote, automated data transmission/operations technology and advanced radiation monitoring technologies could have the following applications: Plant Area Radiation Monitoring, Worker Radiation Monitoring, Effluent Monitoring, Environmental Monitoring (air, water, and groundwater.) Leveraging advanced technologies to risk-inform and automate radiation protection tasks during operation and emergency situations will not only lead to cost savings and improvements in radiation protection and plant operations but also enhance the health and safety of the workers, the public, and the environment.

Additionally, a review of operations and staffing needs for small modular reactors (SMRs) identified several opportunities for automation of typical radiation protection functions. It was determined that a large fraction of the radiation safety functions at a nuclear power could be streamlined with the implementation of more advanced, remote monitoring technologies, application of advanced data analytics and modeling/trending, and utilization of the radiological information to better inform workers, work processes, and reporting needs.

Ultimately, continuous, real time information of the radiological conditions in power plant areas and the environment will enhance worker and public safety. Data from these measurement devices can be accessed at a central operations center and/or displayed around the plant or on handheld devices (smartphones and tablets) as needed by plant personnel. Advanced data analysis tools and intelligent modeling algorithms can be applied to the continuous data to trend and establish predictions of potential changes to the radiological conditions based on plant events or operational changes. This will allow the site to move towards a condition-based radiation protection paradigm where additional monitoring actions are informed by changes in conditions beyond an established baseline, rather than by a pre-established frequency. Automating the gathering of the data along with the application of advanced data trending and modeling tools will simplify the job function of the radiation protection and As Low As Reasonably Achievable (ALARA) organizations. They can then focus on applying the newfound knowledge of the radiological conditions to optimize worker activities and execute on other aspects of radiation protection.

Technological solutions may exist to address pieces of the issue. However, a platform for integrating the radiological information from multiple devices and advanced data analytics with plant processes and systems is currently not available. EPRI is conducting research to identify, develop, demonstrate, and provide information about technologies and strategies that support efficient and safe risk-informed, condition based, and data driven operations.

***Keywords**: Radiation Protection, ALARA, Remote Technologies*

Session 2. ECHO: an innovative device for the measurement of beta contamination in high and fluctuating gamma environments

Bardou Laurent[1], Le Noblet Thibaud[2]*, Maussire Vincent[2]
(Presenting author underlined, * refers to the corresponding author)

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Controlling the radiological cleanliness of environment is a major issue in current radiation protection, whatever the concerned field: security, industrial, environment, health. It requires the ability to carry out controls as close as possible to the worksites at places where the presence of gamma background can be high and fluctuating, disturbing or even making inoperative the devices commonly used (detection limit higher than the cleanliness radiological thresholds, false alarm).

In order to avoid the problems induced by the gamma background, a new probe has been developed and designed by the Carmelec company in collaboration with the CNRS (LabCom P2R): ECHO. Integrating a double sensor and an innovative processing algorithm, this new probe is capable of performing beta contamination measurements in harsh gamma environments.

After a brief reminder of the issue of beta contamination measurement and its associated challenges, the presentation will introduce the new ECHO probe as well as some of its characteristics. A large part of the presentation will be devoted to the performances of the probes. It will present the situations in which the probe can be used as well as the gains brought by this new device.

Fig. 1: photography of ECHO, an innovative device for the measurement of beta contamination in harsh environments

Session 3. Chemical Engineering Perspective on PWR/BWR Colloid Removal to Reduce CRE and Mitigation of No. 1 Seal Failure for PWR Reactor Coolant Pumps

P. Robinson, (n,p) Energy, Inc., Presenter
D. Hultquist, AEP, DC Cook
R. Penney, Xcel Energy, Prairie Island
R. Gerdus, Entergy, Vermont Yankee, (ret'd)

The reduction of radiation source term is an important component to radiation exposure and improve all radiological conditions for radiation workers, while also improving performance on critical safety components. It is simply, sustainable and extremely effective.

What do you need to be successful? Best developed purification technology/media implemented in existing clean-up systems e.g. CVCS and RWCU, and optimized reactor shutdown sequence.

What you don't need? PWRS: Don't you do not need to decontamination CVCS or ultrasonically cleaning of fuel, or use zinc injection.

The focus of this paper is to present key technology solution and report successes at many US Nuclear Power plants. In USA, PWRs are now extending entrance time to RFO by coasting down. This condition risks failure of No. 1 RCP seal. Discussion will include how to eliminate those conditions which fail RCP seal.

To date, this source term solution has been integrated for use during 150 refueling outages (RFOs) in 30 different PWRs reactors and 5 different BWRs. The lead 4 Loop PWR is DC Cook-1,2 which recently completed their 16th refueling outage using this technology solution, saving millions of dollars in O&M costs, and sustainable reduced dose rates.

More US utilities are now adopting the solution, as is international NPP community.

Session 3. ALARA tools: Source Term Control at CNE Cernavoda

Catalina Chitu, Catalina.Chitu@cne.ro, Alexandru Nedelcu, Alexandru.Nedelcu@cne.ro
"CNE Cernavoda" NPP, No. 2, Medgidiei Str. Cernavoda 905200, Romania

Individual and collective dose reduction is one important objective of Cernavoda NPP radiological safety policy. Radiological risks associated with operation and maintenance activities performed in a nuclear power plant must be controlled in a such manner that radiation exposure of personnel be kept ALARA.

Identifying, reducing, and controlling radiation sources are important for both optimizing workers exposure and preventing unplanned exposures.

Also accurate and effective communication of radiological risk is important for source term and personnel exposure control.

Controlling source term at Cernavoda NPP is a real challenge for our organization due to the complex aspects of generation and potential spreading ways.

By controlling systems modifications, the integrity of protection barriers has been maintained in order to maintain radiation fields at levels as low as reasonably achievable.

Significant improvements of main radioactive circuits chemistry control which are in process of implementation will contribute to reducing activation products deposition and, as a result, to gamma radiation fields reduction.

A sophisticated RMS has been successfully used to continuously monitor radiological hazards in accessible areas and in some other zones in order to early detect significant changes or abnormal trends of radiation fields.

Heavy water leaks have been promptly identified by using TAM – tritium in air monitoring system and treated with high priority contributing to significant reduction of personnel internal exposure and environmental emissions.

Outage Activity Transport Monitoring (OATM) surveys permit component radionuclide activities and their radiation field contributions to be trended with reactor operation. These data are required to perform various assessments such as the effects of chemistry changes on radiation fields, evaluation of the source term reduction technologies and decontamination planning.

Defective fuel is identified by continuous monitoring of radioactivity in fuel channels and it is promptly discharged to spent fuel bay.

Good collective dose performances confirmed the efficiency of source term control policy of Cernavoda NPP.

Session 3. Impacts of Ag-110m on Radiation Field Generation: Review of an innovative experiment

Edgar MOLEIRO[1*], Alain ROCHER[1], Thierry TRIBOLLET[2].
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KEYWORDS: Ag-110m, Silver, Radiation field, Radiochemistry

Introduction

Primary coolant silver impurities can be a major problem for PWR plants as Ag-109 (48,2 atom%) from elemental silver can be activated in Ag-110*m* which specific behavior presents difficult challenges for power plant staff. In fact, this radioelement is difficult to monitor, is an important contributor of radiation field of low temperature primary systems. For EDF, as the other operators, a major goal is to find the ways to prevent silver contamination on primary surfaces and effluents and identify the potential actions to take in the event of silver contamination.

Impacts of Ag-110*m* on Radiation Field Generation: Review of an innovative experiment

EDF, as other operators, faces difficult challenges dealing with Ag-110m. For the plants affected by silver contamination this element can be an important contributor to shutdown radiation fields, up to 90% or more for some components, and represent roughly 10 to 15% of outages total exposure exposure[1)2)].,0. Furthermore, the specific chemical behavior of silver makes those events difficult to prevent without disrupt the shutdown schedule[3)]. In order to obtain breakthrough improvements in silver management a recent experiment has been conducted in a French power plant. The purpose of this experiment was to test an innovative disposal that allows to follow in real-time the build-up of Ag-110m deposits and its impact on radiation field generation of some power plant components.

These trial has proved the feasibility of such a disposal and its interest in order to:

- Monitoring silver contamination during shutdown ;
- Optimized shutdown chemistry ;
- Have access to real-time data on local radiation fields ;
- Identify quickly the mitigation options in case of severe silver contamination.

References

1) Frédéric Dacquait et al.,"110mAg Behaviour in PWRs: Lessons Learnt from the EMECC Campaigns," International Conference on Water Chemistry in Nuclear Reactor Systems (NPC 2012), Paris, France, September 2012.
2) Mathieu Habert et al., "Radiological Cleanliness Management of the French PWR Fleet," 21st International Conference on Water Chemistry in Nuclear Reactor Systems (NPC 2018), San Francisco, CA, September 2018.
3) Martin Bachet, Carole Monchy-Leroy, and Jean-Luc Bretelle, "Main Conclusions from the MAI Workshop on Silver 110 Contamination in Pressurized Water Reactors," 20th International Conference on Water Chemistry in Nuclear Reactor Systems (NPC 2016),Brighton, United Kingdom, October 2016.

Session 3. Demonstration of High Purity Germanium Skid

David Perkins[1], J McElrath[1], M Mura[1], D Wells[1], Sara Laning[2], and Leah B Whiteker[2]

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ABSTRACT
Early operating conditions and the lack of online instruments required plant staff to manually enter radiologically controlled areas and collect grab samples. Obtaining and analyzing these grab samples is not only costly and workers collect dose from the sample collection, preparations, and analyses. Furthermore, there are delays between sample collection and data analysis including gamma spectroscopy analysis which rely on a mixture of samples analyzed at various times. In addition to reducing the cost of collecting multiple chemistry analysis and data, the application of automated instruments has the potential to collect more samples per day in real time. The coolant would be analyzed, and then the analytical results were evaluated to the previous result, manually compared to defined limits, and plotted to evaluate for adverse trends more rapidly. It should also be noted that as online instruments are developed and improved, the ability of plant staff to remotely monitor coolant chemistry increased, but even today and in many cases, plant staff are still required to manually collect samples.

As utilities consider changes to plant staffing from a 24/7 coverage to less staff on shift, it is essential that instrumentation provide staff with the confidence in the analytical results and the ability to apply data analytics and aid plant operators in the recognition of an actual adverse condition. The application of modern data analysis tools with these automated instruments allows for a more rapid understanding and performance of system chemistry controls during normal operations, transient conditions, and possibly, the identification of previously missed transients as data historians and inline instrumentation can send data on more frequent basis.

Technology development such as cadmium-zinc-telluride (CZT) technologies provide for a light-weight and an easily implemented technology that is easy to deploy and operate, but these detectors provide for an intermediate energy resolution that makes then considerably better than previous version of NaI detectors, the resolution compared to the high purity germanium systems is not nearly as well defined.

Based on the previous demonstrations with CZT and continued work on other automated methods in the SMART Chemistry program, EPRI proposed the development of an online high purity germanium (HPGe) system with the ability to analyze samples at various frequencies, record the results, and visualize the gamma spectrometric data in real-time as a prerequisite to online coolant gamma isotopic monitoring skid.

This paper provides an overview of the HPGe demonstration at the Monticello Nuclear Power Plant.

***Keywords**: Radiation Protection, ALARA, Remote Technologies, Gamma Spectroscopy, Chemistry Sampling*

Session 5. ALARA in Repair of Reactor Vessel Flange Surface at Krško NPP

Borut Breznik
Krško Nuclear Power Plant

Repair of a reactor vessel (RV) flange means repair by welding, grinding, stoning and polishing – in this case of more than 10 indications on the flange sealing surface. Corrosion pits and other indications could be the result of many years of the reactor operation. They can lead to the damages of the flange face. The flange repair is, therefore, an important maintenance step to help prevent leakages.

ALARA planning on the RV flange repair is presented in this paper with the main focus on workplace shielding and personal protection related to this job. A review of the job tasks is also included together with an analysis of collective dose planning.

Session 5. Case Report of Incident with High Exposure of Radiation

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I want to compete to the ISOE Young RP Professionals Award

Introduction

A routine job-, at Forsmark NPP, changing a filter in the cleaning equipment for the reactor pools resulted in exposure in high dose rate radiation field and unplanned dose to personnel. One of the reasons to the exposure was that a malfunctioning radiation instrument was not able to detect the actual very high dose rate correctly.

Results

The highest obtained radiation dose was well under Sweden's 20 mSv/year limit, thanks to the short exposure time and the swift actions taken by involved personnel once electronic dosimeters and radiation alarms was set of. Moreover, the incident resulted in a highly radioactive filter in a lead box that has to be handled, with led to more unnecessary dose.
On the INES scale this radiological incident has been rated as 1, an anomality.

Lessons learned

A radiation instrument with two detectors one in the low dose rate region and one in the high dose rate region, need to be tested with regards to both detectors before use. In Forsmark it has not been possible to test the high dose rate detector due to the absence of a suitable testing unit with dose rate >10 mSv/h.
Procedures for changing filters in the cleaning equipment for the reactor pools also need to be changed to avoid similar incidents in the future.

Session 5. Radiation protection measures during cleaning of the heat transfer tubes and the bottom of the horizontal steam generator.

Radek Svoboda, Jan Novak (Dukovany NPP), Czech Republic

Visual inspections inside a steam generator showed sediments. In 2021 began the first cleaning of the heat transfer tubes and the bottom of the horizontal steam generator. Cleaning was realized by using remote manipulator. Radiation measures were focused at reducing dose rate and reducing time demands.

Session 6. 3D-CZT Gamma-Ray Spectrometers and Imaging Spectrometers for Source Term Characterization at Palisades Nuclear Plant

David Nestle, RPM Palisades, Entergy,
Meredith Mayers, Willy Kaye, H3D, Inc, Ann Arbor, Michigan

The use of gamma-ray imaging spectrometers at nuclear power plants and other facilities for the purpose of understanding the source term environment has grown sharply in the past 10 years. The primary instrument used at nuclear power plants in the United States is the H-Series unit made by H3D, Inc. The instrument uses CZT detectors, which are able to provide 1% energy resolution along with isotope-specific gamma-ray imaging in a portable, room-temperature system. The system is used at over 75% of nuclear power plants in the United States as well as power plants in Europe and Asia. Primary applications the instrument can be used for include locating primary source term, optimizing and verifying shielding performance, surveys of incoming and outgoing shipments, characterization of HRA and LHRA, and site-wide contamination surveys. In addition to the use of the technology for showing where radiation source term is located, gamma-ray spectroscopy using 3D-CZT detectors can be used to monitor real-time isotopics throughout plant systems. Permanent-mount spectrometer systems have been deployed at several nuclear power plants and have successfully tracked isotopics to provide valuable information to the plant operators during their maintenance outage.

The system can be used to track the isotopics and source term during the forced oxidation chemical decontamination to determine when source term has been reduced enough to allow for continuation of the maintenance activities. These isotopics are also able to track and detect other challenging conditions, such as the contamination of various plant systems with Ag-110m. Additionally, independent testing has been completed through the Nuclear Industry Profieciency Test Exercise in United Kingdom to show comparable results can be achieved through the quantification algorithms used for H3D's CZT as are achieved for HPGe. This paper will discuss lessons learned for this and other applications of gamma-ray imaging technology for source term characterization at nuclear power plants including 7 years of the NATC CZT utility working group study results from the Palisades Nuclear Plant. At Palisades, the charging pump room was found to have significant Ag-110m dose rate resulting up to 85% of the dose field in the cubicle. The successful removal of the Ag-110m contaminate will be explained. Results of the pixelated, 3D CZT measurements of the final CRUD burst at the end of May 2022 prior to permanent scheduled shutdown of the unit will be discussed.

Session 6. ASTRE Development

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Keywords: ASTRE, emergency situations, dose rate

ASTRE (Accessibility Simulation Tool for Radiological Emergency) is a tool dedicated to the evaluation of the accessibility conditions in a nuclear power plant in case of an accident with radiological release in the atmosphere. Its goal is to help the local crisis teams, as well as the FARN (Force d'Action Rapide du Nucléaire, the French dedicated team to operate on the nuclear power plants in case of accident), to evaluate the optimal practical travel path for their operations.
ASTRE is based on a computational fluid dynamics (CFD) database (code_saturne calculations). It describes the atmospheric dispersal of a given source term (either from expert judgement or objective MAAP-DOSE results) given a specified weather forecast. It evaluates at real time the space-time distribution of a plume of reject on the site, as well as the situations in the control room, for the next 4 hours. It will give the on-site agent the dose rate at different points on the power plant, as well as the integrated dose along a given travel path, with the best realistic estimation. Furthermore, it will evaluate the contamination level during the intervention to indicate whether a chemical cartridge respirator is needed.
ASTRE is still being developed, but the paper will describe the different models used in the tool:

- code_saturne calculations,
- calculations of the dose rate from the code_saturne results and the MAAP-DOSE results,
- integration of the weather data in the model,
- as well as the specification of the user interface.

The foreseen developments and improvements will also be presented.

Session 6. Development and Operations of the Ultra-Compact Embeddable Gamma Camera Nanopix

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Remote localization of radioactive hot spots is an important issue for reducing the exposition of operators during nuclear facilities maintenance or decommissioning operations. In order to address this challenge, CEA List has been working since the nineties on the development of gamma cameras, which allow visualization of gamma-emitting spots in the field by the superimposition of gamma-ray images on a visible image of the scene. These developments led to two generations of gamma imagers, industrialized by Mirion Technologies: CARTOGAM, in the mid-nineties and, more recently, GAMPIX (commercialized under the name iPIX) a second-generation gamma imager, based on Timepix pixelated detector, developed in the frame of the Medipix collaboration coordinated by CERN. Over the last years, CEA List worked, in close collaboration with ORANO teams, on an important reduction of size and weight of each building blocks of GAMPIX imager and lead to a twice-smaller and ten-time lighter version of the system, with only 8 cm for a weight of 268 g. This miniaturized version of the system, named Nanopix was widely deployed over ORANO's fuel cycle facilities and its compactness allowed operators to map promiscuous scenes.

Based on the feedback of this highly compact gamma camera, CEA List carried on the developments toward an improved version, embedding a computational capacity close to the sensor and several automation and motorization steps. A specific compact electronic module was designed for connectivity management and power supply. The resulting prototype remains very compact with only 10 cm long and light with less than 450 g and gained a remote operability capability, which is expected to improve the dose reduction for operators. Enhanced with the previously cited improvements, Nanopix miniaturized gamma camera is embeddable on very compact terrestrial or aerial robotic platforms for unmanned interventions in radiologically degraded conditions.

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Session 7. Health Effects from the Chernobyl Accident

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After the nuclear accident of Chernobyl in 1986, the affected populations were exposed to radionuclides accidentally released into the environment that could affect their health. Numerous epidemiological studies have shown the health effects of ionizing radiation in populations exposed to different situations (Japanese atomic bomb survivors of Hiroshima and Nagasaki, uranium miners, populations exposed to indoor radon, etc.). The health of populations affected by radioactive fallout from the Chernobyl accident in Ukraine has also been the subject of epidemiological studies.

The increase in the thyroid cancer frequency is the main demonstrated health effect associated with the radioactive fallout from the Chernobyl accident. The increase in the incidence of thyroid cancer in people exposed during childhood or adolescence at the time of the accident was strong and rapid (three to four years after the accident) and still persists today in these children who have become adults. More recently, from the 2000s, an association has also been observed between exposure to ionizing radiation and the risk of leukemia in liquidators (and children, to be confirmed). Other health effects have been studied, such as solid cancers, cataracts, circulatory diseases or malformations, but the results are not conclusive and need to be confirmed by other studies.

Session 7. Ten years at the Fukushima Daiichi Nuclear Power Station, then and now

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Beginning

At 14:46 on 11 March 2011, a major earthquake of magnitude 9.0 centred on the seabed on the Pacific Ocean side of the Japanese islands occurred, and about 50 minutes later, a tsunami of unprecedented scale hit the Tohoku coast of Japan.

The earthquake caused the Fukushima Daiichi Nuclear Power Station (FDNPS) to lose all external power supply, but emergency power successfully cooled the reactors, and three reactors in operation, Units 1-3, were safely shut down as designed.

However, the tsunami damaged seawater pumps and other outdoor equipment installed on the seaward side for residual heat removal, and the emergency power supply installed underground was lost due to flooding. The loss of these safety-critical functions, such as water injection into the reactors and condition monitoring, caused the water in the RPV of Units 1-3 to evaporate, leading to a severe accident.

Hydrogen generated by the water-Zircalloy reaction between the exposed fuel rod cladding and water vapour accumulated in the reactor buildings, causing hydrogen explosions in Units 1 and 3, and in Unit 4, which was undergoing a periodic inspection, hydrogen flowed from the confluence of the exhaust stacks common to Unit 3 and accumulated, resulting in an explosion.

This was partly due to the fact that the possibility of hydrogen leaking into the reactor buildings in the event of a severe accident had not been envisaged, and no measures had been put in place to prevent hydrogen explosions in the buildings.

The flooding by the tsunami also dealt a heavy blow to the individual dose management system, making some 5,000 electronic dosimeters, charging devices and radiation control systems inoperable. As a result, individual dosimetry was carried out using the approximately 320 electronic dosimeters that remained in each of the buildings in the power station, and the results had to be managed using paper memos. However, due to the overwhelming shortages, the dosimeters were managed so that only one representative person was fitted with it at the same work.

Due to the core damage accident and the explosion of the building, the conventional framework of a controlled area to distinguish it from other locations became meaningless.

The monitoring posts in addition to the exhaust stack radiation monitors did not function due to the loss of power supply, and monitoring cars were dispatched to measure the environment (air dose rates, meteorological data, etc.) at the power station site boundaries and other areas.

The contamination of the entire site due to the large-scale release of radioactive materials and the hydrogen explosion led to an increase in background levels, making it difficult to assess internal doses using the Whole Body Counters (WBCs) that had been installed on the site.

Therefore, a vehicle-mounted WBC was borrowed from the Japan Atomic Energy Agency (JAEA) to be used to assess the doses of workers engaged in emergency work, and the WBC cars were moved to the outside to the low-contamination area for measurement.

In order to respond to the accident, a base other than the anti-seismic building, which was the only one available on the site, was also required, and it was decided to use the J-Village football training facility, located approximately 20 km south of the FDNPS, as a second base, and from 17 March 2011, training for workers involved in emergency work, the wearing of protective equipment and the lending of dosimeters were started there.

Throughout the entire accident, 77 workers exceeded 100 mSv for external exposure (max.199 mSv), 13 workers for internal exposure (max.590 mSv) and 85 workers exceeded 100 mSv in total for external and internal exposure.

Of these, 6 workers exceeded the emergency dose limit of 250 mSv. The staff with the highest exposure dose was 679 mSv (external exposure: 89 mSv, internal exposure: 590 mSv).

All of them were working in the central operation room and were predominantly internally exposed to iodine from the hydrogen explosion in Unit 1 through the emergency door. This door had to be opened halfway due to the drawing in of power lines from the outside following the SBO.

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The total amount of iodine-131 and caesium-137 released into the atmosphere from the FDNPS is estimated to be approximately 10% and 20% of the estimated release during the Chernobyl accident, respectively.

Volatile caesium-137/134 and iodine-131 made up the majority of the dose contribution, in contrast to the Chernobyl accident, where the less volatile strontium, barium and plutonium made up a larger proportion of the released radioactivity.

As in the case of the general public, the most important health effects observed so far in workers are considered to be psychological effects.

No health effects or deaths, such as acute radiation sickness or other deterministic effects caused by radiation exposure, have been identified for the workers engaged in the emergency work. Several studies suggest that there is little risk of hypothyroidism when the absorbed thyroid dose is less than 1 Gy, However, considering that the 13 workers were exposed to high thyroid doses ranging from 2 Gy to 32 Gy, the possibility of developing hypothyroidism in the future cannot be ruled out.

Contaminated water

The biggest problem after the accident was the generation of up to 540 t/day of contaminated water, including the 400 t/day of groundwater flowing into the basement of the reactor buildings and the cooling water for the nuclear fuel debris. This contaminated water was pumped up from the reactor buildings, and after removing caesium and strontium using zeolite adsorption equipment such as SARRY or KURION and reverse osmosis filters, some of the water was recirculated for cooling the nuclear fuel debris, while the rest was stored in contaminated water tanks.

In 2013, the J-Village function returned to the FDNPS and the Advanced Liquid Processing System (ALPS) started operating to remove most of the radionuclides from the contaminated water.

2015 saw the completion of the ALPS purification process for the highly contaminated water in the contaminated water tanks, with some exceptions. In the same year, the sea-side barrier wall and groundwater drain were completed to prevent contaminated water from leaking into the sea, while the groundwater bypass, sub-drain and land-side frozen soil barrier wall to reduce groundwater inflow were completed by 2018.

As a result, contaminated water was reduced to 126 t/day by 2021.

Cesium and strontium have been removed from this contaminated water, which is called "strontium-treated water."

The water from which most of the radionuclides other than tritium have been removed from the strontium-treated water by ALPS is called 'ALPS-treated water'. All these treated waters are stored in a group of 1061 treated water tanks. However, 95% of the total tank capacity of 1.37 million m3 on site is already full, and the dilution and discharge of ALPS treated water containing only tritium is being considered.

The tritium concentration in the discharged water is assessed based on the tritium concentration before discharge and the amount of water used for dilution, and is defined as 1500 Bq/ℓ or less.

This concentration is well below the Japanese Government's safety regulation value of 60,000 Bq/ℓ and WHO guidelines for drinking water quality (10,000 Bq/ℓ), and the immediate annual discharge should be managed not to exceed 22 TBq, which was the discharge control target for FDNPS before the accident.

Dose and radioactivity

Reduction of skyshine doses through removal of debris from the roofs of R/B, and the removal of topsoil, the deep plowing, facing, and other measures in contaminated areas to prevent dispersion and reduce radiation doses have resulted in 96% of the areas on the premises being accessible to workers in general work clothes without protective masks. And the average annual dose to workers has also decreased to 2.3 mSv.
The annual maximum external dose in 2021 was 17.3 mSv, and no significant values have been identified for internal doses. The daily number of workers has gradually decreased from a maximum of 7450 in January 2015 to 3890 as of March 2021.
As of 29 May 2022, the maximum air dose rate from on-site monitoring posts was 1.0 µSv/h and the total β air concentration remained in the 1×10-6Bq/cm3 range.
In seawater monitors, Cs137/134 is below the detection limit and the concentration in total β-water concentration is in the range of 20-30 Bq/ℓ.
Nevertheless, dose rates of 40-80 mSv/h have been observed for example, in the vicinity of the spent fuel pool on the operating floor in the Unit 1 R/B due to unremoved debris, and there are still places on the ground floor where the dose rate reaches 40 mSv/h.

Spent nuclear fuel

Unit 1 has the highest level of contamination, and in preparation for the removal of spent fuel from the pool in FY2027-2028, a large cover will be installed over the entire building, and the removal of debris from the ceiling will be carried out inside the large cover.
In Unit 2, where no hydrogen explosion occurred, based on the results of the operating floor investigation, the construction method was revised so that the upper part of the reactor building is not dismantled and a fuel removal platform is constructed on the south side of the building to access the building and remove the spent fuel in FY2024-2026.
In Unit 3, all nuclear fuel from the spent fuel pool was removed from the building in February 2021, and the risk was significantly reduced.
Unit 4, which was out of operation for routine inspections at the time of the earthquake, had no fuel in the reactor and 1,535 spent fuels were in the spent fuel pool adjacent to the building.
On 15 March 2011, the building exploded due to hydrogen flowing through ducts from Unit 3, but in December 2014, all the fuel in the spent fuel pool was removed from the building and the risk was significantly reduced.
At the time of the accident, Units 5 and 6 were shut down for routine inspections. Even after the tsunami hit, the emergency diesel engines, emergency batteries and power supply vehicles were still operational, and the cold shutdown was possible because the power supply was secured and no major damage was caused. Therefore, the spent fuel pool is currently operating normally.

PCV status

No signs of re-criticality have been observed, as the radioactivity concentration of Xe135 has remained at around 1x10-3 Bq/cm3 in Unit 1 and below 1x10-4 Bq/cm3 in Units 2 and 3.
Temperatures are also stable at 20°C in Unit 1, 32°C in Unit 2 and 25°C in Unit 3.
A remotely-operated vehicle (ROV) was deployed to Unit 1 in February 2022 to investigate the removal of fuel debris, and a molten material, assumed to be nuclear fuel debris, was found and a rise in neutron radiation was detected by the B10 detector on board in the vicinity.

Session 7. The role of the radiological protection expert in stakeholder involvement in the recovery phase of post-nuclear accident situations: Some lessons from the Fukushima-Daiichi NPP accident

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Feedback experiences from Fukushima and Chernobyl situations have clearly shown the importance of involving local stakeholders living in contaminated territories for the rehabilitation of their daily life. In this context, this presentation aims to better address the role of radiological protection experts in the recovery phase of post-nuclear accident situation, in mainly relying on the analysis of local initiatives implemented in the Fukushima Prefecture following March 2011. More particularly, this presentation discussed to which extent radiological protection experts can help local population to address challenges they are facing, through the implementation of co-expertise processes and the associated ethical issues and values they should embody. The last part of this presentation focuses on two current challenges at stake in the Fukushima prefecture and dealing with the dissemination of the co-expertise process to all affected communities, as well as the sustainability of these approaches over time. As a conclusion, the presentation suggests some recommendations in order to better prepare radiological protection experts.

Session 8. Corium stabilization: a challenging Radiation Protection project

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The ‘corium stabilization’ modification aims to increase the nuclear safety level in the case of a reactor vessel failure. This is possible by firstly sealing both the ‘reactor pit’ and the ‘flux measuring instrumentation’ rooms together. Then, it is essential to ensure the spreading of corium and its cooling thanks to the water from the containment spray system. In this way, the corium’s propagation is limited.

The first part of the modification requires improvements on internal and external sumps and drains that must continue to perform their safety functions in contact with corium. Isolation valves have to be replaced with post-accident conditions certified equipment. As some of these valves are placed in the lowest reactor building’s sump, their replacement is unusually complex.

The second part consists in creating hoppers in the wall between the ‘reactor pit’ and the ‘flux measuring instrumentation’ rooms. However, these hoppers have an impact on dose rates, mainly in the ‘flux measuring instrumentation’ room. They must therefore be completed by some specific biological protection.

Each step of the modification implies serious attention from the Radiation Protection department since it requires heavy work within limited space, in a high dose rated environment and with significant levels of contamination. The success of such a project depends on the association with the Radiation Protection department as early as the project preparation phase.

Session 8. ALARA Experience with PWR Thermal Sleeve Replacement at Sizewell B

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Sizewell B is a single 1.2 GW(e) SNUPPS Westinghouse PWR, operational since 1995, with the Reactor Pressure Vessel head replaced in 2006. Although familiar with recent industry operating experience about thermal sleeve wear, the extent of the wear observed during Refuelling Outage 17 in 2021 was unexpected and necessitated the priority replacement of fifteen thermal sleeves. Westinghouse were selected to execute the repairs, with around two month's preparation time from contract placement to work start. The project necessitated multiple entries under the RPV head to cut out the old sleeves and install new compressible replacements, working in a high radiation and contamination environment, with particular concerns about the potential eye doses for workers in areas up to 30 milliSieverts per hour.

This presentation describes the radiological protection controls and experience in keeping whole body and eye dose exposure ALARA. The thermal sleeve project was successfully concluded on time and to around 80% of the original dose estimate of 175 man-milliSieverts, at approximately 9 man-milliSieverts per sleeve, with a maximum individual dose of just under 6 milliSieverts. Almost 700 entries were made under the RPV head, totalling 20 hours of "jump time". Remote monitoring and close co-operation between the thermal sleeve replacement team and the site radiation protection team were pivotal to the safe delivery of the project.

Session 9. ISOE Working Group on Radiological Protection Aspects of Decommissioning Activities at Nuclear Power Plants (WGDECOM) – Outcomes and feedback

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Decommissioning of nuclear power plants is a subject of growing importance for the nuclear industry. During decommissioning of NPPs, economics, technical and organizational challenges are encountered and only few feedbacks are collected and shared at the international level to deal with these challenges.

In November 2014, ISOE (Information System on Occupational Exposure, www.isoe-network.net) decided to establish a new working group dealing with occupational radiological protection during decommissioning activities at NPPs, the WGDECOM. The objective of this working group is to improve the sharing of operational RP data and experience for NPPs in all stages of decommissioning. The WGDECOM focuses on the following items :

- Areas which are the most relevant for effective management of occupational exposure;
- Collection of operational data, in particular collective doses;
- Creation of a network of operational RP experts for decommissioning activities;
- Factors and aspects that play key roles in achieving good RP practices in decommissioning.

Visits have been organized in USA (Illinois and California) and Europe (Sweden, Switzerland, Spain and France). The following topics have been addressed during these meetings and will be presented :

- Regulatory context and strategy of decommissioning,
- Collective doses analyses for high doses works,
- Management of risk of internal exposure,
- Radioactive waste management,
- Integrated risk management.

The outcomes of the last WGDECOM teleconference meetings organized in 2020 and 2021 will also be presented.

Meeting organized in October 2020 focus on defining new Program of Work (PoW) for 2020-2023 period. Main activities identified during this meeting are:

- Conduct technical visits to decommissioning sites
- Maintain networking of WGDECOM experts
- Exchange information between ISOE members
- Explore possibility to create decommissioning exposure DB
- Develop new service
- Cooperate with research reactors in decommissioning
- Collect radiological operating experience
- Cooperate with international bodies
- Provide radiological expertise for NEA Publications

Some technical topics were discussed during the meeting organized in November 2021:

- Overview of IAEA-TECDOC-1954 (April 2021) "Occupational Radiation Protection during the Decommissioning of Nuclear Installations";
- Case studies on the ALARA decommissioning challenges and progress at San Onofre Nuclear Generation Station (SONGS) (USA);
- Use of monetary value of person·Sv in decommissioning.

Session 9. Decommissioning project, management and planning aspects for the decommissioning of Research Reactor Ispra1

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Abstract

On September 26, 2019 Sogin Spa, the state owned Company responsible for the decommissioning of Nuclear Facilities and the Radioactive Waste Management in Italy, took over from the Joint Research Centre (JRC) of Ispra the nuclear facility Ispra1 for its decommissioning.

Ispra 1, the first Italian nuclear installation (Figure 1), is a Research Reactor of the CP5 Argonne type having heavy water as coolant and graphite as moderator and reflector. The fuel was 90% enriched uranium and the maximum thermal power was 5 MW. It was inaugurated on 04/13/1959 and during its operational life, mainly as a neutron source in solid state physics research, the reactor produced 13 500 MWd. After 14 years operation, the reactor was definitively shut down in 1973.

Like other research reactor decommissioning projects, in particular if the end of the operations dates back to more than several decades like in the Ispra1 case, some specific aspects must be taken into account for optimising the management and planning of the activities in terms of economical, environmental and safety aspects, as put into evidence also in IAEA Technical Reports series n. 446. In particular, some aspects of these reactors often differ significantly from other nuclear facilities and must be taken into account, such as:

- A broad spectrum of research reactor types;
- Use of high enriched uranium (HEU);
- Use of many types of fuel, including experimental and exotic fuel
- The broad range and specificity of the experimental work carried out
- The unavailability of detailed historical information
- The necessity of post-operational cleanout for simplifying decommissioning activities.

This article summarizes the program developed by Sogin in the Global Decommissioning Plan for the decommissioning of the Ispra1 Research Reactor which takes into account the said aspects and defines the 3-phase decommissioning strategy:

- **Stage 1**: collection of historical information, radiological characterization of the plant, cleanout and possibly clearance of large and small components used in research experiences, dismantling activities of the systems and components of the primary and secondary cooling circuits;
- **Stage 2**: validation of the activation calculations and evaluation of the levels of contamination for the design and dismantling of the reactor pile and of the horizontal and vertical pits connecting the reactor with the various research installations;
- **Stage 3**: Final Radiological Survey and Final Release.

The decommissioning plan was possible through the analysis of previous radiological contamination data deriving from destructive and non-destructive assays and activation calculations with the MCNP code. These data, which were further analysed in the subsequent development of the decommissioning phases, were used to demonstrate the feasibility and reliability of the planned operations in terms of:

- **contamination / activation management** for waste minimization
- **preliminary dose assessments** to optimize occupational exposures.

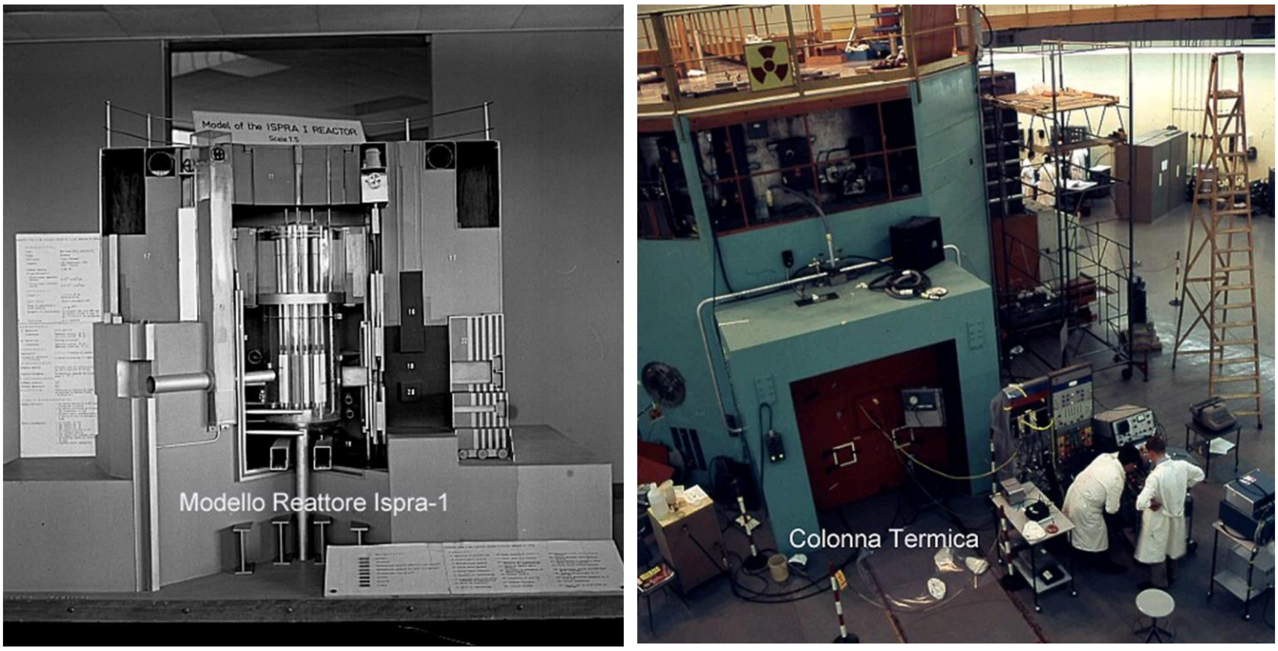

Figure 1 – Chicago Pile n.5 Ispra1

Session 9. Radiological investigations in hard-to-access zones during D&D operations by remote OSL/FO dosimetry

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Abstract: Radiological inspections in hard-to-access zones are achieved with OSL/FO dosimetry, coupled to geolocalization means in order to reconstruct the dose distribution inside infrastructures, sparing heavy duty and large cost in the purpose of setting-up an optimized dismantling strategy.

KEYWORDS: Optically Stimulated Luminescence (OSL), Fiber Optics, Dosimetry, Dismantling & Decommissioning (D&D)

The application of an immediate decommissioning strategy requires the use of innovative dismantling techniques that gradually progress thanks to experience feedback. Among these, remote dosimetry is an essential tool for Dismantling and Decommissioning (D&D) as it helps predict the impact of decontamination procedures in terms of worker exposition and set up cost-effective D&D scenarios.

In open zones, radiological inspections involve gamma cameras and conventional dosimeters (GM, CZT, etc.). Inspections may prove complex, time-consuming and costly in hard-to-access zones, e.g. inside tanks, reactors, storage ponds etc. A particular critical case concerns inspections through small-diameter pipes (< 1 cm) showing small (cm) radius of curvature. Long-range remote dosimetry (20- to 30-meter range or more) in hard-to-access zones is therefore challenging because of signal degradation due to cable length and limitations to miniaturization imposed by the presence of power supply and signal conditioning electronics (sometimes associated with thermal regulation electronics).

In such case, an OSL/FO (Optically Stimulated Luminescence – Fiber Optics) probe is a viable alternative solution, investigated by the CEA LIST since the beginning of the 90's [1], and qualified on three French dismantling sites: Atelier Pilote de Marcoule (APM, CEA/Marcoule, 1998-2000), UP1 fuel factory (AREVA/Marcoule, 2000-2001) and Réacteur Nouvelle Génération (RNG, CEA/Cadarache, 2016) [2]. OSL probes were located at the extremity of armored optical cables, thus robust enough to be pushed and removed within pipes. The main advantages of the OSL/FO technique are Electromagnetic (EM) and Cerenkov immunities, long-range remote operation, high miniaturization, high radiation resistance and wide range in dose rate detection (ranging from 10 µGy/h up to 10 Gy/h) [3-4]. As a passive dosimetry technique, OSL/FO does not require local electronics to operate. The probe head may therefore be made very small (Ø = 5 mm). Finally, it is also waterproof and may be decontaminated.

Until now, radiological profiles were reconstructed step-by-step (curvilinear coordinates). Based on a topographic modeling of the inner infrastructure under investigation, dose reconstruction within every critical parts is usually achieved with the help of Monte-Carlo softwares (e.g. Mercure), sparing heavy duty that would otherwise be necessary to provide access to conventional dosimetric means.

Current work is dedicated to the design of multi-channel (up to 16 channels) OSL/FO fiber cables in order to provide 1D activity profiles. The OSL/FO prototype has also been newly re-designed in a 19-inch/3U enclosure, in collaboration with the SDS company. At the end of the project (2020-2021), the new device will be tested at several D&D sites operated by CEA and ORANO.

This work is conducted within the INSPECT Project, granted by the French government through the PIA initiative (Programme Investissements d'Avenir) and operated by ANDRA (Agence Nationale de gestion des Déchets RAdioactifs).

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Session 9. Development of a Cask for Interim Storage and Final Geological Disposal in Switzerland – Approach, Challenges and Realization

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KEYWORDS: radioactive waste containers, interim storage, deep geological repository, transport

Introduction

Nagra, the National Cooperative for the Disposal of Radioactive Waste, which was established in 1972, is planning the deep geological repositories for low- and intermediate-level waste (L/ILW) as well as for high-level waste (HLW) in Switzerland. Since then, the project has developed considerably and progressed through several phases. Based on a decision of the Federal Council in 2018, the repositories will be constructed in the Opalinus Clay. The Swiss waste acceptance system provides boundary conditions for conditioning waste packages and specifies, for example, acceptable waste packages (limiting e.g. total mass and dimensions).
With the decision in 2013 to shut down NPP Mühleberg at the end of 2019, it became necessary to develop and manufacture containers for decommissioning waste for interim storage and final disposal in the planned L/ILW repository. The project “LC-84/-86” was therefore launched in 2015 by the industry, culminating in the production of a first small-scale series of 30 containers qualified as Type-A in 2020, ready for use in the decommissioning of the NPP Mühleberg.

Design Criteria

The main design criteria were set out in the Swiss specifications for conditioning, interim storage and final disposal, focusing on optimized storage conditions and use of available space in the Zwilag interim storage facility for which potential outer container dimensions were set. In addition, road transport in accordance with the European Agreement concerning the International Carriage of Dangerous Goods by Road (ADR) had to be possible.
Up to 2015, the disposal container concept considered containers with a volume of approximately 14 m3 and 26 m3 with an estimated mass of 60 tons to 80 tons respectively, as these ensured an optimum packaging ratio for different waste types (e.g. decommissioning waste as well as 200-litre drums). In the context of the above-mentioned criteria, the following open points relating to size and mass were identified: difficulty in ensuring safe and efficient handling, especially for decommissioning purposes, Containers not meeting acceptance criteria and storage concepts for the existing interim storage facilities and containers not meeting transport regulations (a Type B qualified container would be necessary for most loading cases), and a transport overpack would have to meet substantial challenges.
A “container working group” made up of members of all the Swiss nuclear power plants, the Zwilag interim storage facility and Nagra as planner of the deep geological repositories was established to develop a new disposal container concept better suited to meeting the main specified design criteria. The main driver regarding the total mass was the criterion for standard road transport (maximum gross weight of 40 tons), resulting in a maximum mass limit of 25 tons per container.
Regarding the optimum use of existing interim storage space and respecting the permitted floor loads for Zwilag, the average gross container mass was set to 20 tons. The main driver regarding the dimensions were Zwilag’s acceptance criteria and existing interim storage concepts. Based on the lattice boxes and their floor grid, the base area was fixed to 8 ft x 6 ft. Respecting the mass limit, while optimizing the filling ratio for the main decommissioning wastes two different heights were finally selected: 4 feet for denser materials (steel-based) – LC-84 – and 6 feet for intermediate-density materials (concrete-based) – LC-86. In this way, both container types share the same large foot area, stabilizing stacks and providing optimum loading areas. The main driver regarding material selection was the repository host rock (Opalinus Clay), which requires

minimization of corrosive gas production. Therefore, it was necessary to develop concrete containers with adequate minimum concrete surplus coverage and minimized reinforcement.

Basic Design Study

The basic design study included the following additional important design criteria: To optimize existing interim storage space, a 10-fold stack had to be considered. The bounding design criteria regarding mechanical loads resulted from consideration of the Zwilag earthquake parameters that the stacks have to be able to handle. Consideration of these parameters resulted in relatively demanding tolerance limits for steelwork and concrete production during project development.

Containers have to be adequately designed from a radiological point of view, and meet ADR requirements. A Type A/IP-II qualification in terms of accident conditions was not mandatory, but meeting the earthquake parameters for interim storage ensured the Type A qualification based on mechanical loads. To ensure efficient packaging and loading, the basic design had to allow a Co-60 activity of at least 2 E+11 Bq (50% of the A2 value as given by ADR). This requirement then defined minimal wall thickness to ensure the dose rate limits set out in the ADR for generalized loading assumptions.

Based on an initial design study and subsequent refinements, two prototypes of the LC-86 were produced as the LC-86 design also covers the requirements for the smaller LC-84 (see Figure 1). Based on these two prototypes, optimization potential was identified, especially with regard to the amount of reinforcement. Additionally, the following points were discussed: the reinforcement concept combined with the steel production design had to be reevaluated as the initial design was impractical and not optimized for serial production, the conditioning of waste (backfilling) was successfully tested, the possibility of decontaminating the as-produced concrete surface was explored and refined calculations for potentially allowed loads were performed based on ADR limits.

Pilot Production and Qualification

As written above, optimization potential was identified based on the two prototypes, especially the required amount of reinforcement steel could be substantially lowered. Additionally, a feasible design for lid fixation as well as a steel band for stabilizing the container top against tensile forces were introduced. Furthermore, different waste conditioning options were taken into account. Swiss interim storage regulations require backfilling of all remaining cavities.

Regarding the envisioned use of the containers, on-site backfilling with the possibility to directly close the concrete container with a compatible concrete lid was considered. Additionally, the use of a centralized conditioning facility was considered, which led to the requirement for a removable and easily manageable lid for raw waste transport. In addition, radiological protection aspects had to be considered and resulted in the development of several different lid options.

The options finally realized cover a raw waste transport lid made of steel, a pre-fabricated concrete lid bolted for final disposal, a pre-fabricated concrete lid to be cast for final disposal as well as a cast concrete lid. The refined design and production specifications resulted in a pilot production of three containers of each type. The participating manufacturers (steel and concrete construction companies, generally from the non-nuclear field) as well as the “container working group” had to continually review the manufacturing process and implement lessons learned whenever possible. The main challenges included tolerances, interface management together with effective quality assurance and finalization of lid options. Based on the successful pilot production, an initial serial and industrial production of a small batch of containers was undertaken. As NPP Mühleberg was shut down in December 2019 and dismantling started immediately afterwards in January 2020, BKW ordered 15 LC-84 and 15 LC-86 containers to be prepared for arising radioactive waste.

Figure 1: Two LC-86 prototypes at Zwilag.

Although the containers were not designed as a Type A package, they were considered to fulfil the requirements due to the layout of a ten-fold stack and meeting Zwilag earthquake requirements, the result being a very robust construction. Final uncertainties regarding their robustness, being constructed of reinforced concrete, could only be laid to rest through tests in accordance with Type A criteria for packages: water spray test, drop test of iron bar, container drop test.

Testing five-fold stacks was skipped as the design criterion specified 10-fold stacks. Testing was conducted on the pilot specimen (see Figure 2). Four different containers were tested in accordance with the above-mentioned criteria, covering the range of raw unconditioned contents and a transfer lid of steel up to a conditioned container with a concrete lid. Masses ranged from 11 tons up to 25 tons.

First, the drop tests were performed in accordance with the height requirements of the ADR from which the containers had to be dropped, depending on the masses. All four containers met the requirements. The same containers used previously were then dropped from twice the required height and even above. All the tests were successful (the containers retained their contents), showing that, during normal transport and in case of accident conditions, no loss of radioactive material was to be expected. The containers could still be stacked on top of each other after the drop tests. Conditioning and especially handling tests at Zwilag were carried out successfully (see Figure 3). The design of the LC-84/86 types is now included in the design criteria for planning and construction of the final conditioning facility in Zwilag.

Overpack Prototype

As the qualification of the container as a Type A transport package was not guaranteed during the design phase, a solution for realizing standard transports was developed in parallel. Based on the dimensions of a 10-ft container, an overpack was designed (see Figure 4) to fulfil the requirements of a Type A package. The requirement of transporting 25 tons (maximum mass of a container) was challenging as the overpack itself had to have a mass of less than 3 tons. This was the only way to ensure that the requirement of a total mass of less than 40 tons for standard transport would be met. A second reason for developing an overpack was the requirement to ensure transport even for potentially damaged containers (in case of unforeseen events during loading and transport in the NPP). Unloading und reloading into another container was not an option, but the existence of an overpack ensures transportability. The following Type A criteria tests for packages were successfully carried out in the production facility of the overpack (see Figure 5 and 5): water spray test, drop test of iron bar and overpack drop test. Testing five-fold stacks was skipped due to the special design of the limited production. During initial handling tests at Zwilag optimization potential was evaluated.

Practical Experience and Outlook

As of today, the first 15 LC-84 and 15 LC-86 containers have been successfully produced and are ready for use as final disposal containers for radioactive waste arising during the dismantling of NPP Mühleberg. Some have already been transported to Zwilag for final conditioning and interim storage.

Figure 2: Drop Test of a LC-86.

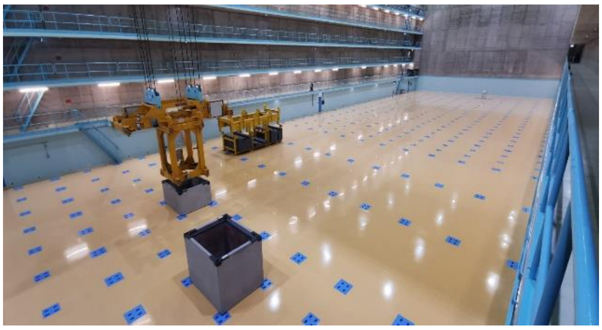
Figure 3: LC-84, LC-86 and storage racks at Zwilag.


Figure 4: Model of the Overpack.

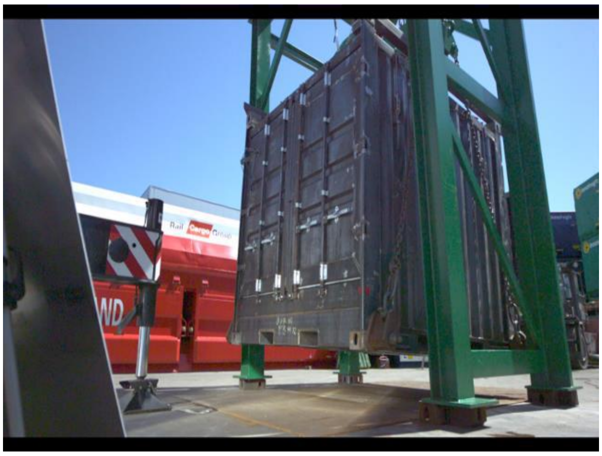
Figure 5: Drop Test of the Overpack.

The small production and handling in practice have shown the following optimization potential:

- Due to stacking requirements, tolerances for dimensions, angles and squareness are small, and compliance is determined by the steel manufacturer. This process therefore needs special attention and quality assurance.
- Cooling the formwork helps to release the container after casting. A greater narrowing angle than the currently used 1° might be beneficial.
- One example of optimized handling equipment is the fixing of a transport rack to the lorry to secure the container in the transport position. This ensures easier handling and subsequently reduces dose to which personnel is exposed.

In the future, new options for these containers might be developed, leading to changes in construction and realization of further optimization potential, especially for handling. As some of these possibilities may impact transport qualifications and the use of these containers for interim storage or in the deep geological repository, the working group will continue to evaluate any foreseeable changes in the future.

Conclusion

The sketched container development is based on criteria and experience during production and handling and provides an overview of the main steps and challenges encountered towards the development of a refined and optimized container design and handling processes.

Today, more and more possibilities for using these containers are foreseen, mainly for saving and optimizing interim storage space and preparing for the ensuing deep geological disposal, i.e. direct storage of the Zwilag coquilles from the plasma torch facility in containers instead of in 200-litre drums.

Session 10. Improving occupational radiation exposure using ALARA tools: performance indicators

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A continued station focus on collective radiation exposure reduction has resulted in top industry performance for CANDU designed reactors over the last 10 years, reducing station dose from 917 man mSv in 2012 to 388 man mSv in 2015.

Many administrative and procedure level measures have been implemented, to improve total collective dose.

The station ALARA committee, led by the Site Director, and the technical ALARA committee continues to provide the strategic direction for achieving consistent low collective dose on both units. Meeting semi-annually and monthly respectfully, these committees provide a critical assessment of performance in meeting ALARA goals and implementing the five-year dose reduction plan initiatives.

The use of working group ALARA coordinators in planning and tracking exposure is contributing in achieving both individual and overall department goals. By improving working group good practices has been obtained excellent results in collective radiation exposure.

Radiation workers, ALARA coordinators, first line supervisors, and managers are directly responsible for controlling and reducing radiation doses. Working group ALARA coordinators have monthly meetings to discuss personnel performance indicators and any other ALARA initiatives to improve radiation protection personnel work practice.

Radiation protection department permanently monitors workers performance inside radiological area. All radiation protection deficiencies are daily analyzed and rapid corrective action are implemented, if necessary. Periodically (quarterly), radiation protection trend analyses are performed to monitor the evolution of the radiation protection deficiencies. 12 categories of deficiencies including: radiation protection work practices (monitoring, contamination control, rad waste collection), contamination control (both personal and material), use of protective equipment, RWP/work planning and ALARA performance indicators have been identified and followed. The most significant improvement has been achieved in controlling of radioactive materials, after implementing of corrective action plan when number of events decreased.

All trend analysis and action generated to improve poor working practice revealed the importance of individual behavior inside radiological area.

Session 10. Dose per RCA-hour; a useful radiological protection indicator?

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The nuclear industry makes wide use of performance indicators to analyse performance and to attempt to understand those aspects of its operation that influence organisational effectiveness. Within Radiological Protection the parameter of dose per hour worked in the RCA is sometimes trended and reported as a measure of dose management performance.

This presentation will review the trends of dose per RCA hour for Sizewell B, NPPs in its sister group, together with a selection of other relevant PWRs, of similar output. It will consider what radiological protection aspects influence this performance measure and judge whether it can be used to provide any useful insights into an NPP's radiological protection programme.

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Poster 1. Radiochromic films for dosimetry

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<u>Abstract</u>

We have developed a passive system of dosimetric films available in gel or powder, which can be used for several levels of sensitivity to UV and gamma radiation.

Legislation imposes limit values for the exposure to ionising radiation for the public and for workers exposed to it. In France, for example, they are 500 mSv for the extrema (fingers).

Today, no other passive product on the market is able to provide such a high detection accuracy. The most sensitive film on the market reacts to 1 Gray, or 1000 milliSievert, which is far from the required values.

In response to the need to visualize low-dose gamma irradiation (from 20 to 500 milliSievert), we have developed a solution that allows workers to directly see a color change informing them of the integrated dose at a given location.

This solution offers several advantages over current solutions:

- Several radiochromic films with different dose level thresholds are available.
- The most sensitive film reacts to 20 milliGray.
- The color change is more visible: it goes from white to blue while existing systems are on a yellow gradient.
- Different formats are possible, which facilitate its integration according to the application: it exists in powder or gel form, and can be shaped on different supports.

Today, we foresee to use the radiochromic films on gloves to rapidly identify when a dose over 20mSv is obtained, and to avoid reaching the maximum dose level.

Poster 2. State of the art and comparison of neutron monitors for measurements on packages loaded with spent fuel assemblies at EDF nuclear power plant.

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ABSTRACT
This contribution presents the recent works on neutron monitors carried out by the French company EDF, in a joint program between the R&D, the production (DPN UNIE) and the nuclear fuel divisions (DCN) with the participation of ORANO TN.
After the loading of spent fuel assemblies in a cask, and prior to shipment, measurements are performed to determine the maximum dose rates around the package. The equipment currently used for neutron monitoring, the PNM200, needs to be replaced due to its obsolescence.
On these grounds the EDF program focuses on the evaluation of the neutron monitors available in the market: the DIAMON (Raylab), the SND (Mirion), the WENDI-II (APVL), the NSN3 (Fuji) and the LB6411 (Berthold). To ensure the fulfilment of EDF requirements, the evaluation rests upon the covered neutron energy range, the gamma rejection rate and the calibration in terms of H*(10).
An extensive experimental campaign has been carried out at different laboratories (ONERA, France), at the ORANO's railway terminal (Valognes, France) and at the TRICASTIN nuclear power plant (Tricastin, France) in order to try and to evaluate the monitors with respect to the reference one, the PNM200 (Mirion), in different experimental conditions.
This presentation will discuss the behavior of the tested monitors in well-known gamma and neutron fields given by laboratory sources (252Cf neutron source, 60Co and 137Cs gamma sources), and in presence of a less well-known mixed gamma and neutron field generated by the spent fuel assemblies. A few of these gamma and neutron fields have been qualitatively characterised using the Diamon, an innovative neutron monitor with spectrometric capabilities, and the Falcon 5000 (Mirion), an HPGe gamma spectrometer.
This contribution will also present the novelty key of these campaigns: the "giraffe", a tailor-made tool for placing and maintaining the detectors (gamma and neutron, also simultaneously) at the measurement point (at contact and at distance of the cask). It will be shown that the "giraffe" greatly simplifies the measurement procedure, and at the same time, enhances data reliability. For instance, it is no longer needed to put up a scaffold to reach the cask's mid-plan, the worker does not affect the measure and the noteworthy weight of the monitor is not an issue any longer. In addition, the "giraffe" offers the remote reading of the measurement on a tablet, finally permitting a significant reduction of the radiation exposure of the worker.
Lastly, advantages and drawbacks of each neutron monitor with particular regard to its use around a spent fuel cask will be dealt with.

Keywords: Neutron monitors, spent fuel, giraffe.

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Poster 3. New workwears for workers in controlled areas

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EDF R&D has launched studies on the design of improved workwears to be used in contaminated controlled areas of nuclear sites in EVEREST mode. As a reminder, EVEREST Project consists in entering the RCA with overalls, suppression of specific RCA clothes. When entering in areas with contamination levels higher than 0.4 Bq.cm-2, workers have to wear specific protective clothes. Also, the studies have focused on protective suits for the workers not to be contaminated during their intervention and during the delicate phase of undressing.

The complexity of the research dictates the implementation of a unique strategy based on an innovative approach of creation and agile development: fast, iterative, interactive and need-based, unprecedentedly applied to radiation protection. The R&D team worked with the workers in order to understand their problems and to propose solutions altogether.

To date, the different prototypes resulting from each step of the method implemented have been tested at the training platforms or in real working sites in controlled area on nuclear power plant. This protective suit has been completely redesigned. It should be noticed that effective characteristics to avoid contamination have been integrated like the innovative opening system which led to a patent. In addition, new features have been integrated (pockets, knee reinforcement, etc.). Moreover, in order to show the effectiveness of this new design, tests with a contaminating powder simulant were carried out.

The last tests will take place in the first half of 2020 and the specifications of this new design of protective suit will be defined and then be industrialized.

Further perspectives are foreseen and aim at integrating into the workwears passive technical innovations to improve the working conditions of workers towards the risk of irradiation.

Finally, this contribution will present the advances made in the design of the protective suits and how the prevention of contamination risks will be improved.

Poster 4. Implementation of Dose Management System for Occupational Radiation Exposed Workers in Ghana

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Abstract
The Radiation Protection Institute (RPI) of the Ghana Atomic Energy Commission used dose management system (DMS) computer software developed by the International Atomic Energy Agency for managing data on occupational exposure to radiation sources and intake of radionuclides. Personnel dose records from the RPI database from 2000 to 2009 were grouped into medical, industrial and education / research sectors. 180 medical facilities were monitored. Highest annual collective dose of 601.2 man mSv was recorded in 2002 and the least of 142.6 man mSv was recorded in 2009 in the medical sector. Average dose per exposed worker for the medical sector was least in radiotherapy and highest in diagnostic radiology, with values 0.14 and 1.05 mSv respectively. Range of average effective doses within diagnostic radiology, radiotherapy and nuclear medicine facilities were 0.328–2.614 mSv, 0.383–0.728 mSv and 0.448–0.695 mSv, respectively. In the education, research and industrial institutions, a total of 34 facilities were monitored. Annual collective doses received by exposed workers in education/research and industrial sectors reduced by ~39 and ~62% respectively, between 2000 and 2009. Average dose per exposed worker for the study period was least in industrial sector and highest in education/research sector with values 0.6 mSv and 3.7 mSv respectively. Range of institutional average effective doses within the education/research and industrial sectors were 0.059–6.029 mSv and 0.110–2.945 mSv respectively. Collective doses in each of the studied categories reduced between 2000 and 2009, an indication that there could be further reduction in subsequent years. This observation could be a result of improvement in radiation protection practices in the respective facilities. Generally, the individual doses also showed reduction with time.

Poster 5. Development of a new probe for beta contamination measurement in a significant gamma background

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KEYWORDS: beta contamination, gamma background

ABSTRACT
Contamination monitoring is a crucial feature for a large range of nuclear applications. The main purpose of this type of measurements consists in guaranteeing that a radiological contamination could not be spread out of a restricted area. The contamination mainly corresponds to alpha or beta contamination: for instance, beta particles emitted by activation products like 60Co or alpha particles emitted by actinides such as uranium and plutonium isotopes. Contamination monitoring is key for controlling the non-contamination at different levels in Nuclear Power Plants (NPP): close to the reactor building or just before the exit of the nuclear site in order to avoid contamination in public areas. In the frame of decommissioning and dismantling operations to be carried out e.g. after the Fukushima catastrophe, the non-contamination control is also of great importance. Finally, in the frame of Homeland Security applications, the control of non-contamination and triage of victims is also a key feature as soon as an accident or a terrorist attack involving a radioactive or nuclear threat could be involved.

Many industrial systems have been developed for several years by the main players of nuclear industry (BERTIN Technologies, NUVIA, Mirion Technologies). However, a current and strong limitation is related to the environmental conditions associated to this type of measurements and especially to the gamma background level. Control in a natural gamma background (close to 100 nSv.h-1 or less) corresponds to reference measurement conditions and do not present any significant issues. Main limitations arise as soon as the gamma background level is increasing: for instance if a non-negligible gamma hot spot is close of the area used for contamination control. In this case, a non-negligible perturbation is induced during the contamination measurement. It corresponds to an additional overcounting (detectors used for alpha or beta contamination detection are usually sensitive to gamma-rays as well) which can occur as soon as the gamma dose rate overcomes 500 nSv.h-1. For a gamma dose rate of a few µSv.h-1, which is quite a standard value in NPP environment, the measurement is not feasible anymore because Minimal Detectable Activity (MDA) would be significantly degraded by the gamma dose rate impact. A device, which would be insensitive to the gamma perturbation, up to dose rates close to several µSv.h-1, would be a technological breakthrough and would make drastically easier the work of practitioners.

In this talk, we will present recent developments made by CEA LIST and NUVIA for carrying out beta contamination measurements in a non-negligible gamma background, up to tens of µSv.h-1. The main focus will be especially made on the development of a handheld probe, including specific plastic scintillator, electronics and specific pulse shape processing. The article will detail the main building blocks included in this handheld system and will expose results obtained during laboratory characterization. First experimental results obtained during in-situ measurements will be also exposed.

Poster 6. Personal monitoring at Czech nuclear power plants and performance of its inspection

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The paper describes the implementation of personal monitoring for the core workers as well as for contractors at the Czech nuclear power plants Dukovany and Temelín. Personal monitoring includes measurement of external and internal exposure of persons and processing and recording of measurement results. Operators of controlled areas must comply with the general requirements of the new nuclear legislation, which has been in force in the Czech Republic since 2017. Compliance with legislative requirements consists in the systematic control of the outputs and results of the inspected party when performing personal monitoring. The paper describes the practical procedures for the performance of the regulator's inspections and the specific facts proving the fulfilment of legislative requirements.

Poster 7. A binocular CeBr3 Compton camera for imaging of weak contamination

M. Z. Hmissi, A. Iltis, C. Tata, G. Zeufack, B. Mehadji, C. Morel and H. Snoussi

Abstract– **To avoid expensive nuclear storage of weakly contaminated material, there is a need for an equipment able to map weak activity contamination. We have been designing in collaboration with the French Agency for Nuclear Waste Management (ANDRA) a Compton camera specially made for that purpose. As the number of gamma rays crossing the detector by second is very low, a key feature of our camera is its high efficiency: valid interactions must not be lost. A second key feature is an exceptional signal-to-noise ratio. To meet these targets, we use two fast scintillating monolithic crystal plates read-out by a digital SiPM. $CeBr_3$ was selected for its high, fast light-output and absence of intrinsic background. For every gamma interaction, the position and energy of the event, including DOI and the relative detection times between the two plates are recorded in both the plates. Our timing resolution below 200 ps allows for a stringent time veto on real Compton events, as they must be recorded simultaneously in both the scattering and the absorber plates. Thus reducing background efficiently. A list-mode MLEM reconstruction algorithm is applied to better estimate the gamma source activity distribution. In this experiment, we have acquired two views of a 30 kBq ^{22}Na source by shifting transversally the camera by 30 cm. Source-distance triangulation allows then to expose a source, even weak, above natural background. We are presently developing a binocular Compton camera for those applications.**

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Key words: Weak activity contamination, Compton camera, gamma ray imaging.

I. INTRODUCTION

Compton cameras based on CZT are starting to be used in the nuclear industry [1]. However, even if they are portable, their sensitivity is limited and their signal-to-noise ratio is too low to efficiently detect weak radioactive sources. Our goal is to realize a Compton camera using scintillator plates that can provide a good angular resolution (< 7° FWHM at 1 MeV), a high efficiency of detection and a very high signal-to-noise. For this, we have chosen $CeBr_3$ as scintillator for its high yield of fast emission and its absence of intrinsic background radiation. We are also using a time gating < 300 ps between the scattering and the absorber plates to exclude non-coincident events. To get high quality interaction positions, we take advantage of "Temporal Imaging", a new concept for gamma ray imaging, which exploits both the scintillation photon spatial distribution and their time of arrival distribution in monolithic scintillators [2]. The camera is portable and fully integrated. In this paper, we present an experiment of binocular imaging of a weak (30 kBq) source at large distance (3 m) using this Compton camera concept which, and demonstrate the interest of binocular observation.

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II. MATERIALS AND METHODS

The Compton camera Temporal consists of two monolithic scintillator crystals. The scattering plate, located in front of the source, is a (32 × 32 × 5) mm^3 $CeBr_3$ crystal. It is followed by an absorber plate, which is a (32 × 32 × 12) mm^3 $CeBr_3$ crystal. Each scintillating crystal is coupled optically to a Philips Digital Photon Counter tile, DPC-3200-22 sensor. The distance between the two plates is 27 mm. The Compton camera is integrated in a system whose mass is < 4 kg that includes acquisition and processing electronics, detector cooling and power supply (figure 1).

Fig. 1. Portable Compton Camera Temporal V2.2.1

The energy resolution of both the plates was evaluated for 511 keV gamma rays: it amounts to 8 % FWHM for the scattering plate and 7 % FWHM for the absorber plate.

A coincidence time resolution (CTR) of 180 ps FWHM without DOI corrections was measured by placing a ^{22}Na source between the 2 plates. The positions of the events in both plates were calculated using time corrected light distributions. The spatial resolution (X,Y) was 1 mm FWHM for the scattering plate and 1.5 mm FWHM for the absorber plate.

Our algorithms record for every gamma ray interaction the position and energy of the event in both the plates, including depth-of-interaction (DOI), even for the thin one, and the relative detection times between the two plates.

Monte Carlo simulation is used and compared to experimental measurements to evaluate the efficiency of our Compton imaging system. It was determined for a 0.5 MBq ^{22}Na source located at coordinates (0,0,500) mm from the center of the front face of the camera. Such a position was chosen to test the performance of the data acquisition and processing electronics with a high impinging gamma ray flux.

The efficiency of our Compton camera design determined by Monte Carlo simulations amounts to 4.2 ‰ and the experimental measurement gives 3.7 ‰.

III. RESULTS AND DISCUSSION

Our ^{22}Na source is a disc of diameter 25 mm with an activity of 30 kBq. It is placed on a pillar of concrete located 300 cm away from the camera front face.

The ^{22}Na source reconstruction is shown on a 100 x 100 image representing the field of view of the camera corresponding to the forward hemisphere. Two acquisitions were performed for 15 hours each for the two camera positions shifted transversally by 30 cm as shown in figure 2:

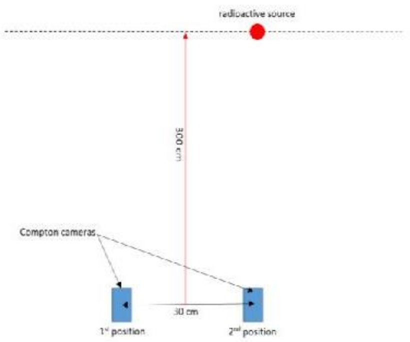

Fig. 2. Description of the experimental setup

For both the positions, our camera could detect the weak source and verify that the observed spot was reconstructed at the right position within the Compton image. Figure 3 shows the two Compton images that were obtained at 511 keV.

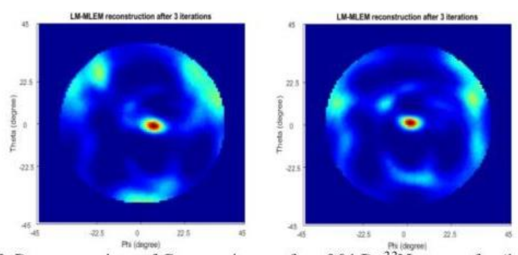

Fig. 3. Reconstructions of Compton images for a 30 kBq ^{22}Na source for (left) the first position and (right) the second position of the camera.

The spherical coordinates (θ,φ) of the source direction are (–1.140°,5.710°) for the first position of the camera and (1.145°,0.146°) for the second position. Knowing that the camera has been shifted transversally by 30 cm, we have estimated the distance of the source from the parallax between the two source directions and have found a distance (3.08 ± 0.21) m FWHM from the front face of the camera.

As it can be seen in bot the reconstructed images, the artefacts are not located at the same positions. Hence, by requiring a distance range for the source, it is possible to exclude all the artefacts and get a clean and reliable detection of this weak source.

The binocular version of the Temporal Compton camera presently under-development is presented below (figure 4). With such a camera design, we can detect a 30 kBq ^{22}Na source within less than 3 h.

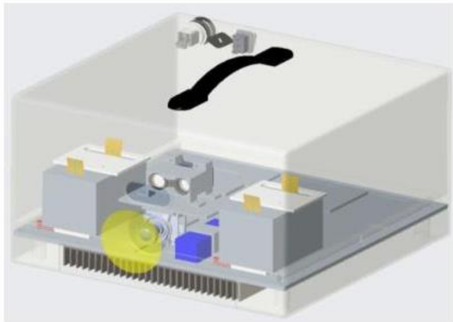

Fig. 4. Binocular Compton Camera design under development

IV. CONCLUSION

Our target was to develop a portable Compton camera able to image weak contamination with a good angular resolution. Such a camera will be very useful for radioactive waste classification and can even image natural radioactive background.

In order to test low activity imaging, a ^{22}Na source was located 300 cm away from the camera front face. The acquisition lasted for 15 h for two positions of the camera shifted transversally by 30 cm. Image reconstruction was performed after selecting the 511 keV annihilation peak of the ^{22}Na energy spectrum. 320 Compton events were detected for the first position and 329 for the second position. The source was detected on both the images. The distance of the source measured by using the parallax between the two Compton images is ~3 m. This parallax effect allows also to further exclude random detection and represents a key for low activity imaging.

We are presently developing a binocular Compton camera with four plates similar to those described above.

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Poster 8. 3D-CZT for Gamma-Ray Isotopic Characterization and Quantification at Nuclear Power Plants

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Health physics and radiation protection professionals at nuclear facilities benefit from accurate measurements of the identities, locations, and quantities of radioactive isotopes. Often these measurements must be done in situ, where cryogenically cooled detectors are not convenient to deploy. Three-dimensional position-sensitive pixelated CdZnTe detectors hold the world record energy resolution for detectors operated at room temperature, so they are a natural fit to applications in such environments. Current fieldable systems are able to achieve 1% FWHM at 662 keV and use either Compton Imaging or mask-anti mask coded aperture imaging to provide source localization information. Next-generation prototypes that are under development can achieve as low as 0.5% FWHM at 662 keV. Since CZT operates at room temperature, this has allowed for the development of instruments that are better suited to provide isotopic characterization and quantification in the field. This paper presents the latest advancements of this cutting edge pixelated CdZnTe technology, several instruments that use this technology, and example measurements in relevant situations.

H-series imaging spectrometers have been in use in over 70% of the US nuclear-power fleet and at many nuclear facilities around the world since their introduction in 2013. The system integrates the three-dimensional position-sensitive CdZnTe detectors, associated readout electronics, an embedded computer, a battery with over 7-hour life, an optical camera, and a laser range finder in a portable waterproof enclosure. The S100, a spectrometer, applies the excellent energy resolution and efficiency of the pixelated CdZnTe platform in order to quantify and show the time evolution of the radiation intensity and the source activity for each isotope. This instrument has been successfully deployed at several commercial nuclear power plants. The quantification results are validated by comparing to the conventional reactor-coolant chemistry sampling and subsequent counting with laboratory HPGe detectors. The comparison study will be presented. The A400, a hand held radioisotope identification device (RIID), uses the same underlying CZT technology so it has the same excellent energy resolution and great efficiency, in addition to directionality without the need for cryogenic cooling. All of these cutting-edge capabilities are packaged in a compact and economic handheld design of less than 2.3 kilograms.

Quantification algorithms have been developed for use with H3D's 3D-CZT systems. A radiation transport model has been combined with a detector efficiency characterization toolkit that is unique to each product model. A collimated option is available for scenarios when accurate isotopic quantification is required in high background environments or in the presence of other sources of the same isotope. The radiation transport model is capable of handling a cylindrical or rectangular geometry; currently options for a pipe, a drum or a box are available with more advanced geometries under development. The latest results from the testing of these quantification algorithms will be presented.
Results from the measurements collected at commercial nuclear power plants and other applicable sites will be presented to show how these instruments can be used to detect, identify, localize, quantify, and show trends from gamma-ray sources.

Poster 9. Automated Primary Coolant Gamma Isotopic Analysis module for the EPRI SMART Chemistry Project

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Measurement of gamma isotopic activity in the primary coolant of an operating Nuclear Power Plant [NPP] is a routine operations task. These results are used to monitor and trend for fuel reliability and asset management. Today this task is primarily performed by manually extracting samples of the fluid in the plant, taking them to the assay laboratory, dispensing the proper amount into the assay container, and then counting on a HPGe gamma spectroscopy system. This process is time consuming for the NPP staff, creates risk of spills of radioactive fluid, and exposes the staff to radiation fields. Furthermore, potentially useful information from short-lived radionuclides and dynamic activity changes between the discrete sampling periods is lost.
EPRI decided to incorporate an inline gamma isotopic analysis module in its SMART Chemistry system to enable automatic and real-time gamma isotope monitoring and trending of the primary coolant.

The design used here is an evolution from an earlier EPRI project using a shielded and collimated CZT detector continuous spectroscopy system aimed at an existing pipe containing primary coolant within the plant [1]. That detector and shield was very convenient for quick and low-dose installation, and worked very satisfactorily for the intended purpose of determining the major nuclide components of the external radiation field. This project adds an improved resolution detector to better detect the minor constituents in the primary coolant mix, and to allow a wider dynamic range in activity. The shielding was also improved to allow more freedom in placement within the facility. A special feature was added to allow both continuous assay of the non-decayed fluid, and then to periodically assay an aliquot of the decayed fluid.

This new HPGe version is stand-alone external module as shown in Figure 1. It receives the reduced pressure and temperature primary coolant from the plant, either directly from the plant, or from the SMART Chemistry sample conditioning skid. The module is comprised of a coolant handling component, a detection system, and a spectral acquisition and analysis and data handling platform.

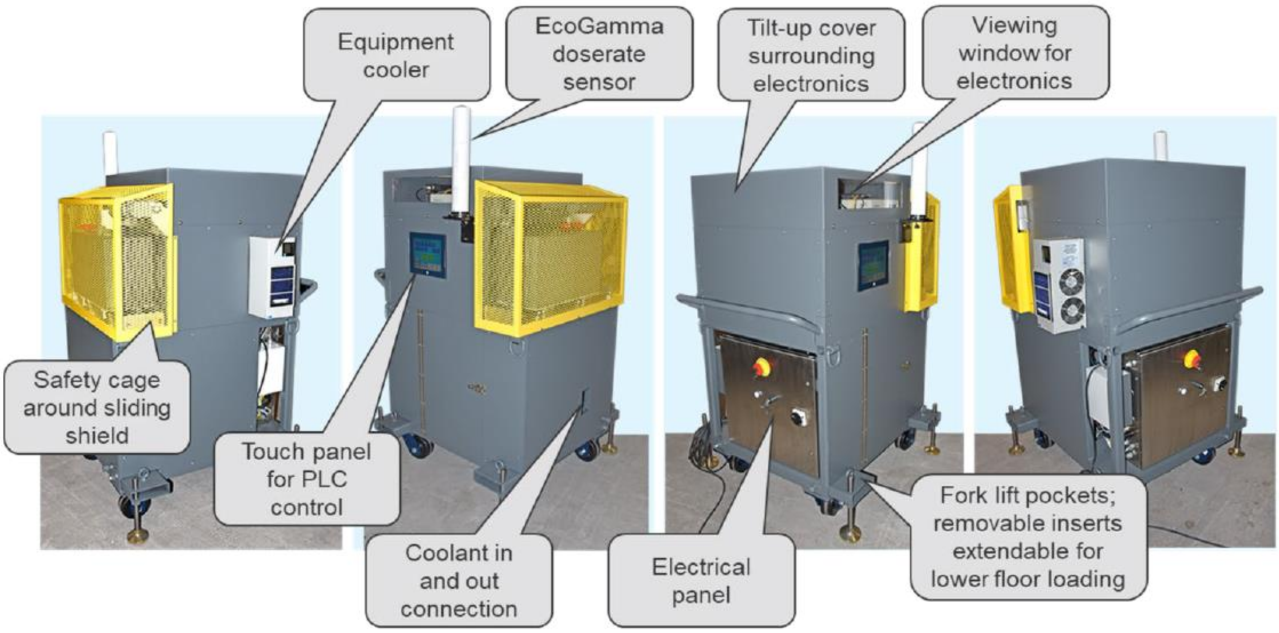

Figure 1 The EPRI Primary Coolant System as seen from all sides.

The detection system is a shielded HPGe detector, optimized for a wide dynamic counting range and wide energy response range. The detector is a Broad Energy detector of modest size, model BE2825. The shield is compatible with much larger detectors and large sample chambers for lower detection levels, or much smaller detectors and smaller sample chambers for higher activity fluids. The detector is cooled by a model CP5+ high reliability electrical cooler. To allow operation in elevated environmental temperatures up to 50degC, the housing around the cooler is insulated and a small air-conditioner is attached to the housing. See Figure 2.

Figure 2 Detector cooler inserted into the shield, and the air conditioner. The access panel has been removed.

A unique aspect of the system is the ability to perform a continuously repeating sequence of assays on a continuously flowing sample, or to extract a sample and institute a decay period before counting, or to continuously alternate between both types of measurements. In the alternating mode, for example the system would normally do continuous assays, then once every 24 hours would count a sample that has decayed for 24 hours, then return to the continuous mode. This feature is accomplished by having two separate detection chambers. The liquid capacity of the chambers can vary between about 2ml to 450ml; chambers of 2, 8, 133 and 450ml were supplied. The small chambers are used to continuously assay flowing liquid with no decay. The large chambers are to sample and hold the fluid, and then do an assay after decay. Figure 3 shows this aspect of the shield and sample chambers.

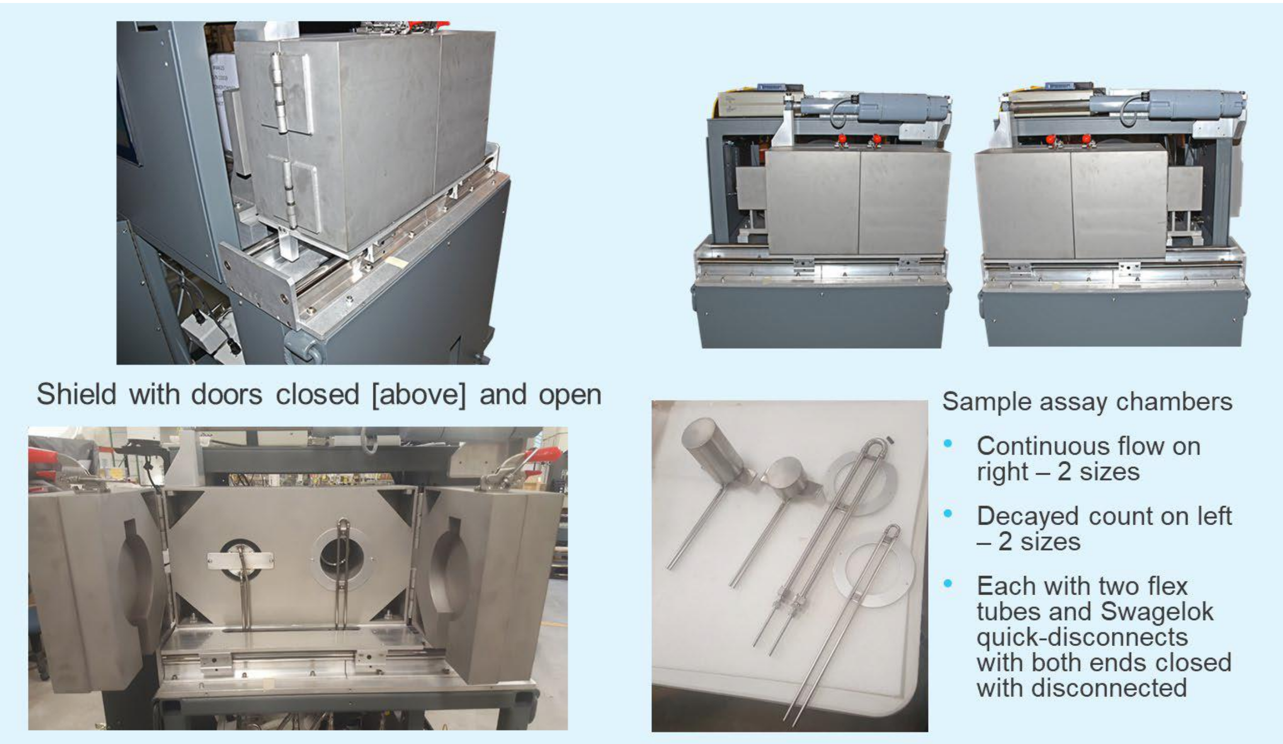

Figure 3 Shield with sliding dual assay chambers

Another very important feature is a built-in Quality Control source. A small natural Thorium source is embedded into the stationary part of the shield, and continuously exposed to the detector. The source is attenuated by several cm of tungsten to remove the low energy gamma lines where they could cause interference, while preferentially retaining the high energy lines that are easily separated by the detector and software. Thus each spectrum has a built-in QC check proving proper operation. These QC data are also separately tracked and displayed by the Horizon database and display software.

Another unique aspect of the system is the ability to simultaneously execute analyses with multiple counting times on each sample; short count times to record dynamic changes in activity, in parallel with long count times for lower detection limits of minor constituents. A typical setting would be to record a series of 30 minute assays, 3 hour assays, and 24 hour assays for the continuous flowing chamber. This capability is provided by the Data Analyst, a new Mirion product. The Data Analyst can control HPGe detectors [in this case] but it can also control Scintillation detectors or CZT detectors. All acquired spectra are analyzed using the standard Genie2000 algorithms which are running inside the Data Analyst. Results are tested for high nuclide activity and doserate. Any alarms generate a signal which can be used to control external actions, such as turn on lights and bells or turn off valves.

The top of the module has a hinged cover that when open exposes the electronics chamber. This area contains the Data Analyst, the Lynx MCA, and the CP5+ cooler controller. Figure 4 shows these items with the hinged cover removed, and with the yellow safety cage around the sliding shield assemble removed.

Figure 4 Electronics area, shown with top cover removed and safety cage removed.

The coolant handling system in the bottom cavity of the module automatically controls the filling process for the assay chambers, monitors and records flow and temperature, controls the dual sample assay process, and controls the process to flush the chambers and to measure background and chamber contamination. Incorporated into the design are many safety features. These include leak sensors, over-pressure relief values, over-temperature flow shutoff valves, back-flow limiting check valves, and dual solenoid flow control valves at critical locations. These items are shown in Figure 5. All the processes are controlled by a PLC with a touch-panel interface. The control screen is mounted on the front surface of the module as seen in Figure 4. The PLC main menu for the coolant handling portion of the system is shown in Figure 5.

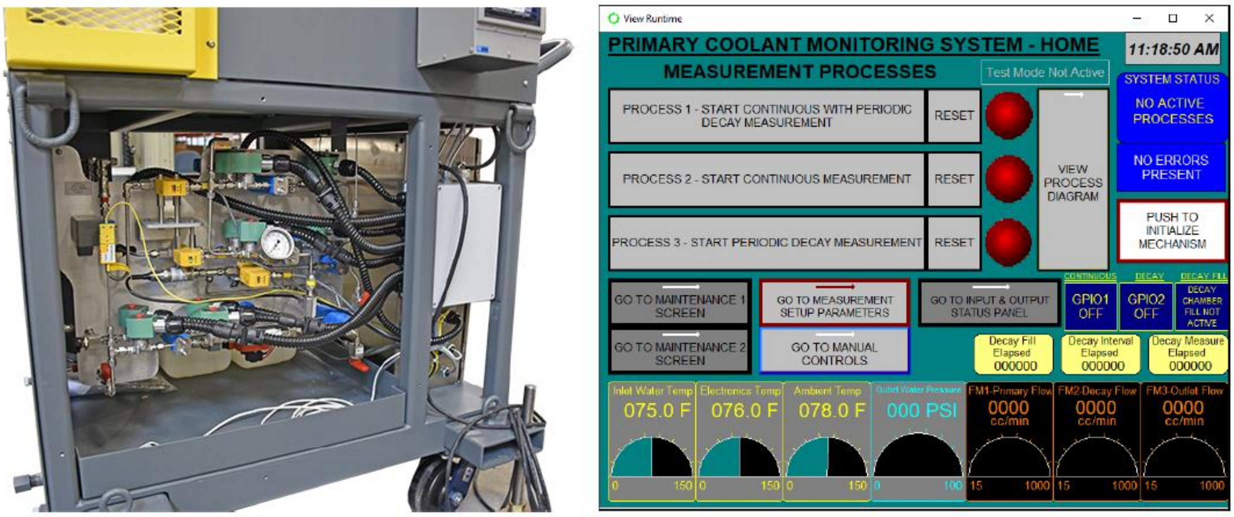

Figure 5 Coolant handing system shown on the left is in the lower portion of the module. The PLC primary control screen is on the right.

All of the spectral data and doserate data generated by the system is stored within the Data Analyst. It has the capacity for several years of primary data and results. As each datapoint is generated it is transmitted over an Ethernet link to the remote Horizon display and database. This is the primary user interface. Figure 6 shows the most commonly used screen. All raw data and spectral results are also stored in the Horizon computer, as they are generated. If the data link between the two units is broken, the remote display will stop updating and indicate an error; when the connection is restored, the Horizon system will immediately resume the display, and simultaneously will start the process to reach back and restore the missing data.

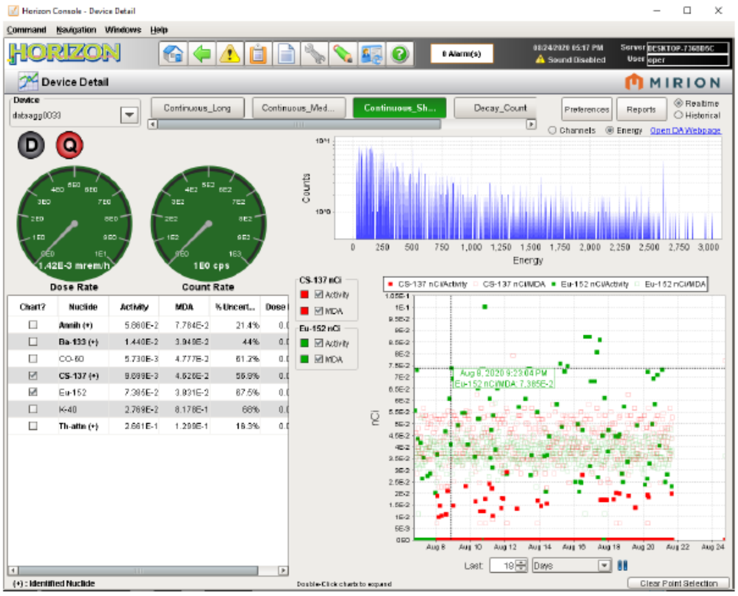

Figure 6 The most commonly used Horizon remote user display screen. All the nuclides in the library would be listed and can be displayed by ticking that nuclide.

Figure 7 shows the key performance parameters of Minimum Detectable Concentration and Maximum Concentration. It also shows various physical and environmental parameters.

Construction of the system and Factory Acceptance testing was completed in December 2020. Monticello Nuclear Generating plant was selected by EPRI as the initial Demonstration site. David Perkins of EPRI was the lead. The system operated from late-January 2021 through mid-August 2021. The operation was very reliable – there were no unplanned shutdowns or lost data. Over 6500 gamma spectra were generated and automatically analyzed.

- Inlet water temperature: up to 40deg C
- Inlet water pressure: >210 kPa, <350 kPa
- Inlet flow rate: 20 – 200 cc/min
- Power: ~1000 VA
- UPS: 12 hour
- Environmental temperature: 50deg C with AC on
- Weight: 1200 kg for skid, 270 kg for battery rack
- Floor loading: 35-80 kg/m^2, depending upon setup
- Minimum Detectable Concentration – Bq/ml
  - Shutdown: 0.05 to 0.5; OnLine: 0.05 to 40
  - Large Decay Chamber: 0.0005 to 0.005
- Maximum Concentration - ~2 E6 Bq/ml

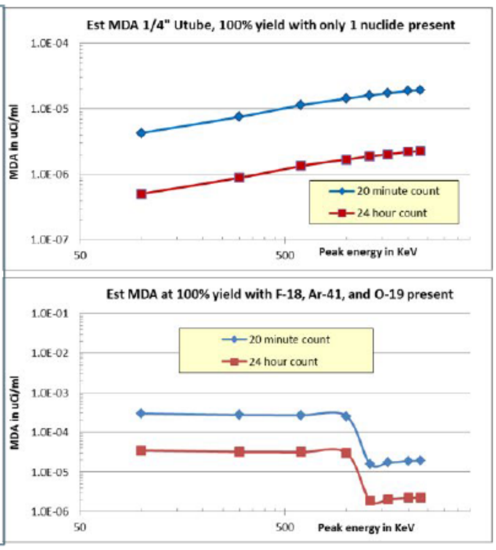

Figure 7 Key performance parameters, capabilities, and limitations of the Primary Coolant Monitor.

All of this happened under rather severe COVID-19 restrictions at Mirion, Monticello, and EPRI. Travel was restricted and site access was limited. Virtual acceptance testing and training was done, which went quite well, or perhaps better as it allowed more people at each location to participate. There was Excellent cooperation by the Monticello staff, which included allowing remote access for Mirion personnel to view and even control the system when installed and operating in the plant, after following very specific access control protocols. This was very effective at allowing observation and adjustments to be quickly and easily made.

Keywords*: Primary Coolant, Automated Gamma Spectroscopy, HPGe spectral analysis*

References:
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Poster 10. Chemical decontamination of RHRS and/or CVCS circuits, one of the main drivers of EDF's ALARA approach

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KEYWORDS: Decontamination, RHRS, CVCS, Dose rate, Radiochemistry

Since 2004, EDF has been implementing chemical decontamination with the aim of reducing the radiation fields of the most contaminated units to the average for their series. The deployment of an innovative methodology, based on radiological diagnosis of the units (primary and auxiliary circuits) and on the planning of heavy maintenance operations on circuits that are candidates for decontamination (RHRS and / or CVCS), now allows for the construction of a multi-year unit remediation program for the Fleet. Typically, 3 to 4 units are cleaned up per year. The characterization of the radioelements in the circuits to be decontaminated, made by CZT gamma spectrometry, determines the choice of the redox chemical solution to implement. Decontamination processes have been developed and qualified in order to effectively dissolve the most harmful radioelements in doses integrated by personnel such as Co60 and / or Ag110m.

This publication describes the state of play relating to the organization and implementation of the decontamination operations of the auxiliary circuits of the EDF Fleet. Furthermore, it draws up a comprehensive review of the feedback from the decontamination units since 2016, taking into account the types of deposits to be removed, the chemical solutions implemented, the effluents generated as well as the potential impacts of the decontamination operation on the critical path of the shutdown. This communication presents the results obtained in terms of the post-decontamination dose reduction factor and overall gamma activity eliminated. This feedback also integrates the combination of qualified processes sometimes implemented in order to save time on the operation schedule as well as reducing effluents. In order to reduce the volume and activity of the effluents produced, it also includes the treatment by ion exchange resins when necessary. In addition, this publication opens ways to optimize the process by measuring the concentration of chemical and radiochemical elements in the various oxydizing and reducing phases of the process. Finally, the text presents investigations regarding the kinetic behavior of circuit recontamination.

Poster 11. Advanced training in the field of radiological Characterisation for clearance / radioactive waste

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Radiological characterisation and clearance require specific knowledge and skills. Targeted advanced training of the staff is one possibility to get this knowledge and skills. As decommissioning is more and more a challenge in the future, the knowledge how to reduce the amount of radioactive waste by clearance is getting more important.

A modular training was established, which allows adaption to specific requests of the institution and state of knowledge and skills of participants. The modules have following focuses: “Radiological Characterisation and Clearance” and “Sampling - Methods and Execution”. The results can additionally be used for characterisation of radioactive waste.

Participants working in groups apply new knowledge to examples of the installation or simulated scenarios. Afterwards they present and justify the results. This concept was carried out for different institutions, like Nuclear Power Plants and research reactors as well as accelerators and specific legacies.
Several training workshops were performed to improve the knowledge how to clear installations, buildings, areas and legacy waste. This special knowledge is usually not present during the operational phase of the nuclear facility but is essential for decommissioning.

Workshops for radiological characterisation and clearance were already performed in Bangkok and Hanoi (organised by IAEA), in Switzerland (CERN, Paul Scherrer Institute PSI, PSI Education Center) and in Germany (NPP Grafenrheinfeld). Training courses in sampling had been organised several times in German NPPs Brunsbüttel, Krümmel and Rheinsberg.

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