ISOE European Symposium

Uppsala, 26 – 28 June 2018

Programme & Book of Abstracts

Supported by

The European Technical Centre of the Information System on Occupational Exposure (ISOE) is pleased to organize, in collaboration with and the support of Vattenfall and the Swedish Radiation Safety Authority (SSM), the 2018 ISOE European Symposium on Occupational Exposure Management at Nuclear Facilities.

The Symposium will be held in Uppsala, Sweden, from the 26^{th} to the 28^{th} of June 2018. 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 25^{th} of June 2018, two meetings devoted to specific audiences have been organised:

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

Finally, a technical visit has been organised to Forsmark NPP site on Friday 29^{th} of June 2018 that will give you the opportunity to see the SKB depository for low and intermediate waste, the mobile logistic center intended for emergency situations as well as a to visit unit 3 controlled area.

We are looking forward to welcoming you in Uppsala,

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

Programme Committee Members

Lucie D'ASCENZO	ISOE ETC, CEPN – France
Laure-Anne BELTRAMI	ISOE ETC, CEPN – France
Borut BREZNIK	Krško NPP – Slovenia
Aude GENIAUX	ASN – France
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Caroline SCHIEBER	ISOE ETC, CEPN – France
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:

Uppsala Konsert & Kongress (UKK)
Vaksala Square 1
753 31 Uppsala

Entrance from Vaksala Square and Storgatan

ISOE European Symposium

Uppsala, 26-28 June 2018

PROGRAMME

TUESDAY 26 JUNE 2018

08:30 - 09:00	*Registration*

09:00 - 09:30	Opening Ceremony C. Schieber (Head of ISOE ETC), L. Berglund (Deputy CEO for Forsmarks Kraftgrupp AB), F. Hassel (Deputy Director General of SSM)

Session 1. **Chairpersons**	**RP Programs** **G. Ingham (ONR, UK), V. Simionov (Cernavoda NPP, Romania)**
09:30 - 09:50	Radiation Protection and ALARA Program Highlights at Ontario Power Generation J. Zic (Pickering NPP, Canada) *Distinguished paper at 2017 Fort Lauderdale ISOE Symposium*
09:50 - 10:10	The Monetary Value of the man.Sievert (Alpha Value): Outcomes of an ISOE Survey S. Andresz, C. Schieber (CEPN, France), T. Jobert (EDF DIPNN Direction Technique, France)
10:10 - 10:30	Dose Reduction and Target Settings at Oskarshamn NPP C. Bauréus Koch, M. Pettersson (Oskarshamn NPP, Sweden)

10:30 - 11:20	*Coffee-break, Visit of Exhibition, Posters*

Session 2. **Chairpersons**	**Source-Term Management (1/2)** **A. Ritter (Leibstadt NPP, Switzerland), M. Johansson (Ringhals NPP, Sweden)**
11:20 - 11:40	Source Term Successes and Challenges at LaSalle County J. Moser (LaSalle NPP, USA) *Distinguished paper at 2018 Fort Lauderdale ISOE Symposium* Presented by M. Wakeley (LaSalle NPP, USA)
11:40 - 12:00	EDF's Radiation Protection Strategy for an Optimized RHRS and CVCS Circuit Decontamination Plan A. Rocher, J. Jaubert (EDF GPEX, France)
12:00 - 12:20	PWR and BWR Source Term Reduction: A Chemical Engineering Perspective P. Robinson ((n,p) Energy, Inc, USA)

12:30 - 14:00	*Lunch Break*

Session 3. Chairpersons	**Source-Term Management (2/2)** A. Ritter (Leibstadt NPP, Switzerland), M. Johansson (Ringhals NPP, Sweden)
14:00 - 14:20	Overview of International RCP Operation Practices during Shutdown - Use of ISOE Forum and Contacts LA. Beltrami (CEPN, France), T. Jobert (EDF DIPNN Direction Technique, France), L. Vaillant (CEPN, France)
14:20 - 14:40	BWR System Surface Contamination: Comparison of Three Similar Units with Different Development M. Olsson, L. Ekerljung (Forsmark NPP, Sweden)
14:40 - 15:00	Reactor Coolant System Monitoring using Portable Cadmium Zinc Telluride Detectors B. Boyer (Prairie Island, USA)

15:00 - 15:50	*Coffee-break, Visit of Exhibition, Posters*

Session 4. Chairpersons	**RP at Decommissioning Stage** S.G. Jahn (ENSI, Switzerland), G. Ranchoux (EDF DP2D, France)
15:50 - 16:20	ISOE Working Group on Radiological Protection Aspects of Decommissioning Activities at Nuclear Power Plants (WGDECOM) I. Calavia (CSN, Spain)
16:20 - 16:40	Preliminary Design of the RPVs and Internals Dismantling of the Trino and Caorso NPPs. Effective Dose Optimization E. Amoroso, M. Caldarella, F. Mancini, R. Botti, D. Annunziata (SOGIN, Italy)
16:40 - 17:00	Simulation of the Occupational Radiation Dose caused by Contamination of Primary Circuit Media in Pressurized Water Reactors in overall Maintenance and Refuelling Inspections and Decommissioning Work S. Schneider, A. Artmann, G. Bruhn (GRS, Germany)

WEDNESDAY 27 JUNE 2018

Session 5. Chairpersons	RP Regulations: Guidelines and Implementation P. Weickert (EDF GPEX, France), P. Hansson (SSM, Sweden)
09:00 - 09:20	UK Regulatory Approach to ALARA in Light Water Reactors at the Design Stage V. Rees (ONR, UK)
09:20 - 09:40	Design, Modification and Maintenance of Contamination Barriers in RCA as a Challenging Part of Radiation Protection in Nuclear Facilities S.G. Jahn (ENSI, Switzerland)
09:40 - 10:00	Status of Readiness for Lens Dose Limit Change M. Johansson (Ringhals NPP, Sweden), V. Nilsson (Forsmark NPP, Sweden)
10:00 - 10:20	The Brief Introduction of Eye-lens Radiation Dose Survey in Chinese NPPs Q. Cao, L. Liu, X. Wei, R. Zhao, W. Xiong, Y. Xiao, S. Xia (China Institute for Radiation Protection, China)

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

Session 6. Chairpersons	Accident Management A. Geniaux (ASN, France), J. Ma (IAEA)
11:10 - 11:30	Operational Radiological Protection Regulatory Treatment applied to the Implementation of the Containment Filtered Venting System and the Alternative Emergency Management Center P. Diaz Arocas, M. L. Rosales, J. Peña, T. Labarta (CSN, Spain)
11:30 - 11:50	Ringhals Evacuation Centre (REC) - a Mobile Contamination Scanning Unit for Accident Situations A. Ljungberg (Ringhals NPP, Sweden)
11:50 - 12:10	Lessons Learned from the Analysis of the Return of Populations following the Lifting of Orders of Evacuation after the Fukushima Accident T. Schneider, P. Crouail (CEPN, France)

12:10 - 12:30	Conclusions from the Radiation Protection Managers and the Regulatory Body Meetings

12:30 - 14:00	*Lunch Break*

Session 7. Chairpersons	RP Indicators S. Hennigor (Forsmark NPP, Sweden), T. Stahl (GRS, Germany)
14:00 - 14:20	Performance Monitoring of Radiological Safety Policy at Cernavoda NPP V. Simionov, L.A. Samson, I. Popescu (Cernavoda NPP, Romania)
14:20 - 14:40	Analysis on Occupational Exposure of Radiation Workers in Korea based on KISOE Database (2007~2016) B. Kim (KINS, Republic of Korea)
14:40 - 15:00	How to Use the ISOE Database? L. D'Ascenzo (CEPN, France)

15:00 - 15:50	*Coffee-break, Visit of Exhibition, Posters*

Session 8. Chairpersons	Contamination Management (1/2) G. Renn (Sizewell B NPP, UK), B. Breznik (Krško NPP, Slovenia)
15:50 - 16:10	Management of Alpha Emitters in the RCS at EDF and Comparison with International Utility Approaches - Presentation of the Data collected from an ISOE Questionnaire T. Jobert, G. Penessot (EDF DIPNN Direction Technique, France), C. Dinse (EDF CEIDRE, France), L. Vaillant, S. Andresz (CEPN, France)
16:10 - 16:30	Organization to Fight against Workers Internal Alpha Contamination in Decommissioning Works at Saint-Laurent A J. Laurent (EDF DP2D, France), B. Boussetta (EDF DIPDE, France), G. Ranchoux (EDF DP2D, France)
16:30 - 16:50	Internal Dose Assessments at Forsmark NPP L. Ekerljung (Forsmark NPP, Sweden)

19:00	**Symposium Dinner at Norrlands Nation** Address: Västra Ågatan 14, 753 09 Uppsala	

THURSDAY 28 JUNE 2018

09:00 - 09:15	Technical Visit Information – S. Hennigor (Forsmark NPP, Sweden)

Session 9.	**Contamination Management (2/2)**
Chairpersons	**G. Renn (Sizewell B NPP, UK), B. Breznik (Krško NPP, Slovenia)**
09:15 - 09:35	New Beta Probe for Contamination Monitor under a Gamma Background K. Bourdergui, V. Kondrasovs, M. Hamel (CEA List, France), B. Feret, D. Rothan (NUVIA, France), C. Dehe-Pittance, R. Woo (CEA List, France)
09:35 - 09:55	ALARA & Management of Internal Exposures at CNE Cernavoda C. Chitu, I. Popescu (Cernavoda NPP, Romania)

Session 10.	**Job Experiences (1/2)**
Chairpersons	**F. Klímek (Temelin NPP, Czech Republic), I. Calavia (CSN, Spain)**
09:55 - 10:15	A New Application for Industrial Radiography M. Petit (EDF Lab Paris-Saclay, France), P. Weickert (EDF UNIE-GPEX, France), N. Filliard, O. Petitprez (EDF Lab Paris-Saclay, France)
10:15 - 10:35	ALARA Experience with PWR Baffle Bolt Replacement at Cook D. Wood (Cook NPP, USA) *Distinguished paper at 2018 Fort Lauderdale ISOE Symposium*

10:35 - 11:25	*Coffee-break, Visit of Exhibition, Posters*

Session 11.	**Job Experiences (2/2)**
Chairpersons	**F. Klímek (Temelin NPP, Czech Republic), I. Calavia (CSN, Spain)**
11:25 - 11:45	Spent Fuel Storage Rack Replacement J. Bonnefon, P. Audry, F. Josse, G. Saulnier, E. Gas Nocera (EDF DIPDE, France), G. Sacre, B. Kopecky (REEL, France), C. Comas, D. Damelet (NUVIA, France)
11:45 - 12:05	Operational Experience of the first Dry Fuel Storage Campaign at Sizewell B NPP R. Parlone (Sizewell B NPP, UK)
12:05 - 12:25	US Dry Cask Dose Comparisons D. Miller (Cook NPP, USA)

12:25 - 12:45	Distinguished Papers and Closure of the Symposium

FRIDAY 29 JUNE 2018

Technical Visit (optional)

A technical visit is organised (full day) to the **Forsmark NPP** with:

- SKB depository for low and intermediate level waste
- Mobile Logistic Center, intended for emergency situations
- Short tour to Forsmark unit 3 controlled area, where you can look the reactor hall, turbine hall and main control room through windows.

The bus departures and arrivals will be announced later on the ISOE Network website and to the registered participants.

The participation is subject to availability as the visit is restricted to a limited number of persons.

BOOK OF ABSTRACTS

ISOE European Symposium
Uppsala, 26-28 June 2018

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ORAL PRESENTATIONS

POSTER PRESENTATIONS

Session 1. The Monetary Value of the man.Sievert (Alpha Value): Outcomes of an ISOE Survey

Sylvain Andresz[1], Caroline Schieber[1], Thomas Jobert[2]

[(1)] Nuclear Protection Evaluation Centre (CEPN), Fontenay-aux-Roses, France
sylvain.andresz@cepn.asso.fr

[(2)] EDF Direction Technique de la DIPNN, Lyon, France

The principle of optimisation in radiological protection aims to reduce radiation exposure as low as reasonably achievable (“ALARA”), taking into account economic and societal factors. To guide and facilitate the application of this principle, the International Commission for Radiological Protection (ICRP) has proposed to put into balance the costs associated with radiation protective actions that may be implemented to perform a task with the benefits in terms of reduction of exposure [1]. ICRP later suggested to use cost-benefit or cost-effectiveness analysis, where the benefits (or the effectiveness) are expressed in a monetary unit and linked with a monetary reference value of the avoided unit of exposure: the monetary value of the man.Sievert, often called the “alpha value” [2].

In 1997, the ISOE European Technical Centre (ISOE ETC) performed a first international survey among nuclear regulatory authorities and nuclear utilities, not only to gather the alpha values but also to collect information about the rationales used to evaluate it as well as its statute, role and use in the country [3]. A second survey [4], then a third one [5] were later performed in 2003 and 2009 to update the data and the information.

Almost 10 years after the last survey and following a request from EDF/SEPTEN, it appeared advisable to investigate the evolution of alpha values. Therefore ISOE ETC performed a survey among ISOE participants during Spring 2017. The survey has been concretely implemented using questionnaires sent by email to specific ISOE contact persons. One questionnaire was designed for nuclear regulatory authorities and one for nuclear utilities/NPPs. Answers were received from 21 countries. A synthesis of the answers can be found in ISOE European Technical Centre Information Sheet No. 61 [6].

The updated alpha values (or system of values) collected by the survey will be presented and put into comparison. The presentation will also provide information and discussion about the statutes and the uses of the alpha values. Some perspectives about new methodologies to assess the statistical monetary value of the human life, sometimes used as a basis to determine the alpha value, will also be presented.

Session 1. Dose Reduction and Target Settings at Oskarshamn NPP

C. Bauréus Koch, M. Pettersson
Health Physics and Dosimetry Department, Oskarshamn NPP, Sweden
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Mobile: +46 76 125 03 51

Oskarshamn NPP has long been using planning values for individual doses and dose targets for collective doses for each unit. Dose budgets are used for activities within the units and departments in order to reduce doses. A process has started to challenge the different ways of reducing dose. This includes looking at new targets for different groups of workers, limiting the number of persons in the controlled area and reducing dose for the recurring work in operation and during outages.

The new targets and the work on how they were set will be presented.

Session 2. EDF's Radioprotection Strategy for an Optimized RHRS and CVCS Circuit Decontamination Plan

Alain Rocher, Jérôme Jaubert
EDF/UNIE, France

In the French PWRs, more than 90% of collective doses are due to activated corrosion products deposited on the out-of-flux surfaces. Almost 80% of these doses are integrated during the shutdown phases. The goal is to lower both individual and collective doses to an-achievable level as low as possible (ALARA). To reach this objective, it is more than important to reduce the contamination by corrosion products and to optimize the radiological monitoring, which is a challenge for EDF's fleet of 58 reactors. This optimized monitoring of the primary and auxiliary contaminations circuits is based on 3 indexes: Reactor Coolant System (RCS) index, Reactor Building index and gamma spectrometry CZT (Cadmium-Zinc-Tellurium) measurements. Checking these indexes leads to the optimal decontamination program presented in this paper.

- RCS index: This index is calculated from the dose rate measurements around the primary pipes. It represents mostly the state of the loop contamination and the global primary system itself. The location of the measurement points is perfectly identified and the measurements have to be taken just at the beginning of the reactor shutdown in order to avoid misinterpretation. This interpretation index is essential as it has existed since the beginning of the EDF fleet operation.
- Reactor Building index: This index has been specified since 2011 to evaluate the auxiliary circuit contamination and to early identify an over-contamination. It is measured at the beginning of the shutdown before the coolant oxygenation. Sub-indexes can be calculated to evaluate the state of a particular circuit if necessary.
- CZT gamma spectrometry measurements: These measurements describe the activated corrosion product contribution to the dose rates at different circuit locations. It could permit an early detection of specific contamination and could help to choose the appropriate decontamination process, depending on the nature and the half-life of the radionuclide (Co60, Co58, Ag110m, etc.).

This presentation deals with the dose rate results and analyses the complete approach of characterization of the contamination. It focuses on the most polluted plants to be cleaned-up, in order to restore their radiological state to the series average. The aim of the new strategic approach is to determin the optimal RHRS and CVCS circuit decontamination plan. Currently, the robust index and the maintenance program after decontamination are taken into account. These two aspects are weighed and the result is integrated into the series matrix. Then, a multi-annual program for decontamination is suggested, 3 or 4 units per year. Finally, the decontamination process will be implemented under the management control of the Nuclear Power Plant.

Session 3. Overview of international RCP operation practices during shutdown – Use of ISOE forum and contacts

Laure-Anne Beltrami[1] (laure-anne.beltrami@cepn.asso.fr); Thomas Jobert[2] (thomas.jobert@edf.fr); Ludovic Vaillant[1] (ludovic.vaillant@cepn.asso.fr)

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[2] EDF DIPNN Direction Technique, 19 rue Pierre BOURDEIX 69007 Lyon, France

For the EDF fleet, RCP are operating during forced oxidation and the subsequent purification stage, one of the criteria considered to shutdown the last RCP is a 'residual' activity of 50 GBq/t of Co-58 (the activity at the time of the shutdown is often lower than this value).

EDF would like to compare EDF practices concerning RCP operation during shutdown and their radiological impacts with international practices. In order to do this, EDF has collected feedback experiences regarding the existing RCP shutdown strategies and their basis/rationales by using a questionnaire sent to some NPPs over the world. The ISOE network has been used to identify the contacts to whom this questionnaire was sent. Ten answers of the questionnaire were received and analysed to compare practices used, criteria selected and its origin.

This presentation will address the following:

- Context of the EDF study,
- Presentation of the questionnaire use for this request,
- EDF practices,
- Analysis of the questionnaires received

Session 3. BWR System Surface Contamination: Comparison of Three Similar Units with Different Development

Mattias Olsson and Lina Ekerljung

Forsmarks Kraftgrupp AB, Chemistry and Radiochemistry, SE-742 03 Östhammar, Sweden
Phone: +46(0)173-810 00, +46(0)173-819 52; E-mail: mso@forsmark.vattenfall.se

Nuclide specific measurements are done annually to estimate the surface contamination inside selected pipes and heat exchangers at Forsmark NPP. The purpose is to complement the dose rate measurement program with a more detailed analysis of selected measurement locations. The measurements show which nuclides that contribute to the dose rate. Also, the relation between nuclides of different half-lives can give information on the stability of the surface oxide layer or how prone a system surface is to adsorb contaminants. The measurement program in its original form was devised by the reactor vendor Asea Atom in the 1970s and has been developed and kept active at Forsmark to provide continuous trending to this day. An overview and lessons learned presentation that covered the first 30 years of surface activity measurements was held at the 2012 ISOE European Symposium in Prague: *BWR System Surface Contamination: Three Decades of Nuclide Specific Measurements*.

In recent years a number of modifications and other actions have been done at the three BWR units at Forsmark NPP. At the same time, some notable changes have occurred for the surface activity. The most dramatic example of such a development is shown in Figure 1: since 2012, the contamination and dose rates of parts of the systems connected to the reactor have increased at unit 2. It is beneficial to have comprehensive data for three similar units as the development is analyzed, and surface contamination and chemistry data for all three units have been compared to support conclusions for the reason behind the increase at unit 2. It is clear that a system decontamination in combination with an elevated concentration of Co-60 in the reactor coolant is a central driver for the increased dose rates.

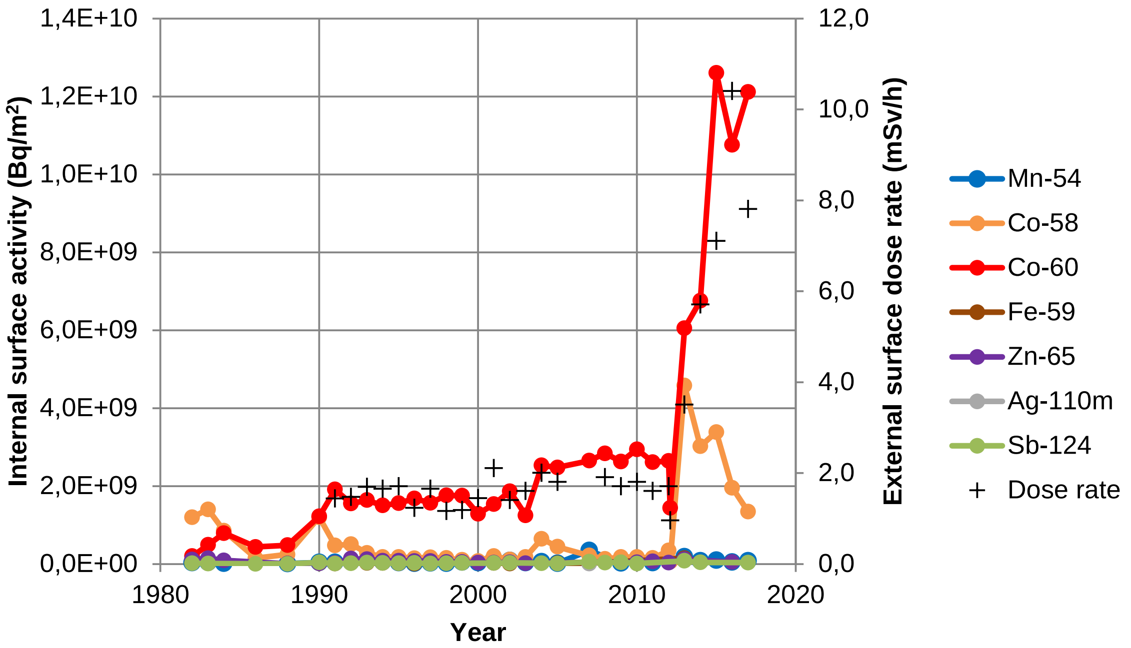

Figure 1. Trends of internal surface contamination and external surface dose rate for an RHR pipe at Forsmark 2, 2002–2017.

Specific events that have been considered and that could potentially have affected surface activity trends are:

- System decontaminations that have been done at units 2 and 3,
- the exchange of internal reactor parts such as core shroud lids at units 1 and 2,
- the introduction of new fuel models with larger Nickel base alloy surfaces,
- the introduction of forward pumping of high pressure drainage at unit 2 – approximately 11 % of the feed water flow now bypass the condensate polishing plant, and
- a thermal power uprate from 2928 MW to 3253 MW at unit 2.

It is important to investigate the root cause behind the increase of surface contamination at unit 2 thoroughly as it could affect the dose to the staff. This will be done further; the current state is included in this presentation while additional conclusions will be submitted to a future forum.

Session 3. Reactor Coolant System Monitoring using Portable Cadmium Zinc Telluride Detectors

Brad Boyer, Prairie Island RPM
brad.boyer@xenuclear.com

Prairie Island Nuclear Generating Plant has implemented an innovative reactor coolant system monitoring system that has reduced refueling outage costs by providing real-time characterization of hard gamma emitting radionuclide concentrations. The system utilizes several cadmium zinc telluride spectroscopic process monitors to continuously characterize the system in-situ. This allows for enhanced projection of post forced oxidation purification times, real-time verification of the purification system functionality, and improved prediction of collective radiation exposure impacts from radionuclide concentrations.

Session 4. Preliminary Design of the RPVS and Internals Dismantling of the Trino and Caorso NPPs. Effective Dose Optimization

*Elena Amoroso** (amoroso@sogin.it), *Massimiliano Caldarella** (caldarella@sogin.it), *Francesco Mancini** (fmancini@sogin.it), *Renzo Botti*** (botti@sogin.it), *Daniela Annunziata**** (annunziata@sogin.it)

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*** Sogin Trino NPP, Strada Statale 31bis del Monferrato, 13039 Trino (VC), Italy

SOGIN is the National Company in charge of dismantling of Nuclear Facilities and Radioactive waste management in Italy. Sogin currently initiating the dismantling of the "E. Fermi" Trino Nuclear Power Plant (NPP) and Caorso NPP, respectively a Pressurized Water Reactor (PWR) facility and a Boiling Water Reactor (BWR) facility, in particular the preliminary design of the different phase for dismantling of Trino's and Caorso's Reactor Pressure Vessel (RPV) and Internals will start.

From the Radiation Protection (RP) side, the dismantling of a facility presents a series of characteristics that set it aside from normal operation, such as for example the continuous variation of the type and level of the radiological and conventional risks and the fact that activities are performed on equipment and systems that have not been acted on previously. In addition, certain of the protection systems are left out of service and have to be replaced with mobile systems. It is also important to point out that many of the workers participating in dismantling projects are not accustomed to this type of work and may even not have worked before with ionising radiations and therefore the optimization about occupational exposure in the design phase is necessary in order to carry out a good training.

This article describes the radiological protection plan implemented through dose evaluations and exposure scenarios using Visiplan 3D ALARA Nuclear Codes for the activity of the preliminary design phase about dismantling of RPV and Internals.
The study covering all aspect of practical implementation of radiation protection for Occupational Exposure Management in terms of dose evaluation to workers for its performance during every scenario in normal conditions.
Finally, the results obtained are analysed in terms of effective dose and collective dose.

The preliminary design of the dismantling phase I for each of the two plants was developed for the following activities:

- ***the dismantling of the RPV Internals and Radiological Vessel Characterization of Trino NPP*** **(Figure 1)**. In particular, the effective collective dose for vessel opening activities, handling internals and management of samples taken, is equal to about 18 mSv * man for about 600 h * man. It has also been estimated that the most exposed worker will take approximately 6.25 mSv for the entire duration of the activity.
- ***the dismantling of RPV of Caorso NPP*** **(Figure 2)**. As part of the preliminary design of the dismantling activities of the Caorso RPV, after the removal of the Internals, we estimated the effective dose to optimize the choice of cutting techniques.
  In particular, 2 operating conditions were considered: RPV without shielding (water); RPV partially filled with water and in the process of being dismantled with a "mixed" technique, i.e. keeping the water level just below the cutting plane of the spools.

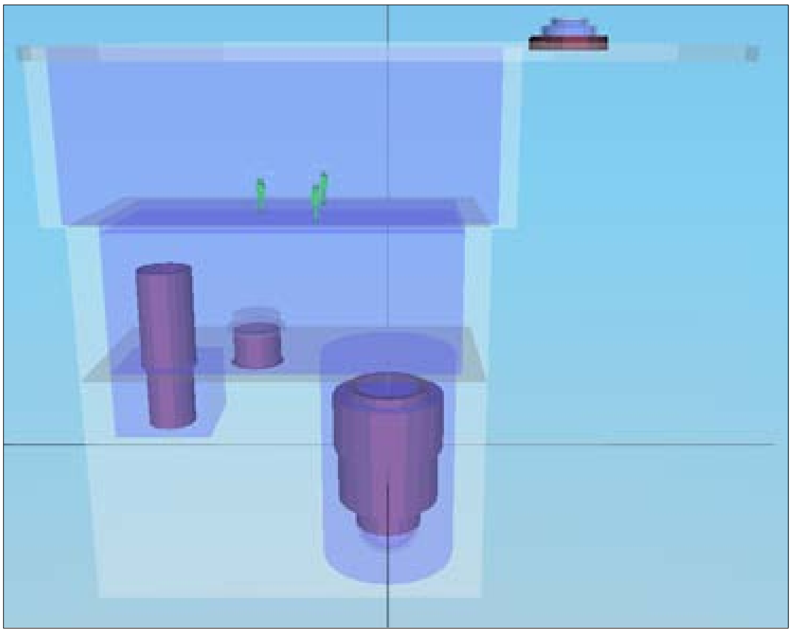

Figure 1- Dose Assessment - Dismantling Internals and Radiological Characterization of RPV Trino NPP

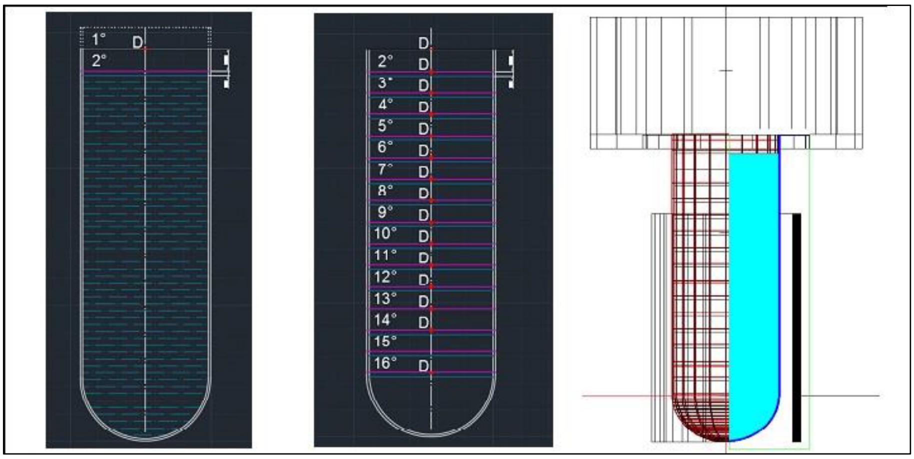

Figure 2- Dose Assessment – Dismantling of RPV Caorso NPP

Session 4. Simulation of the Occupational Radiation Dose caused by Contamination of Primary Circuit Media in Pressurized Water Reactors in overall maintenance and refuelling inspections and decommissioning work

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1 Introduction

The occupational radiation exposure of workers in nuclear power plants during overall maintenance, refuelling inspections and decommissioning is not only determined by the activation and contamination of structural elements of the primary circuit, but also by a number of additional parameters such as the geometry of shielding, self-shielding of components, deposits of radionuclides, the planning of tasks within the controlled area or even the behaviour of workers. The reduction of radiation exposure by activated components can only occur by using shielding, whereas the radiation exposure caused by contamination of components may be also minimized by the chemical operation mode of reactors as well as by application of systematic decontamination techniques. Influencing the contamination within the primary circuit insofar offers the greater potential for improving radiation protection. The number of parameters influencing the occupational radiation exposure leads to a complex problem which is addressed by a comprehensive generic model.

2 Basic Information

Technical literature of the past three decades was analysed to gather existing knowledge about the formation and deposition of contamination within the primary circuit. The currently available approaches for modelling water chemistry and the transport of radionuclides in the primary circuit are presented in recapitulation e.g. in [1,2]. It was shown that in most cases the high complexity of chemical and thermodynamic processes within the primary circuit requires a large number of modelling parameters. On the other hand, there are only few data on actual local dose rates available. Simulations with many degrees of freedom and few measurement data to fit to the simulation generally lead to increased uncertainties. Therefore the existing models tend to be rather specific to individual facilities – thus not generic – and do not lead to general deductions. In [2] it is concluded that a step back to simpler models might be more expedient.

3 Modelling

Data on the concentration of radionuclides dissolved in the primary coolant of the past 15 years were analysed. The results serve as a basis for the compilation of generic nuclide vectors. For the determination of nuclide vectors to describe contamination caused by deposits, only metal oxides are considered [3]. The generic 3D model of the primary circuit presented here is based on the analysis of technical documentation of German nuclear power plants. With the aid of engineering drawings of Siemens/KWU-NPPs with pressurized water reactors of the 1200 MW+ power class, a 3D CAD model was constructed (Fig. 1). It contains the relevant components from the point of view of radiation protection. With it, the spatial relations during the performance of tasks can be well illustrated and investigated. With that one can decide which systems and components act as a relevant source and in what cases shielding has to be considered.

Tasks are modelled as a combination of retention times and positions in the surroundings of work areas, so that mean personal doses of the personnel can be determined. The coordinates of relevant positions are mathematically transformed in a way that they are suitable as input for the MicroShield software [5]. Using this, the contribution of any subcomponent and subsystem to the local dose rate is calculated and summarised afterwards. There are extensive data sets of actually measured local dose rates, but they are limited to a few positions (steam generator water chambers and main coolant pump outside hot/cold) and points in time (overall maintenance and refuelling inspection). Additional measurement data are collected by concrete enquiries, e.g. during visits to facilities. The ISOE database provides information about actually accumulated personal doses due to tasks (mainly) during overall maintenance and refuelling inspections and about the related amount of work in man-hours. Information on occupational doses and effort during decommissioning works is available in annual reports of operators, providing a lower level of detail than (operational) ISOE database values. Based on this information a job model for dismantling of steam generators and reactor coolant pumps was created that allows predictions of the related dose expectations.

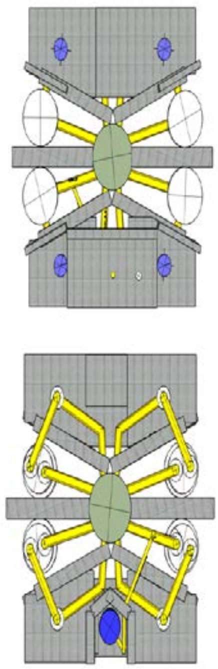

Fig. 1 Views of the 3D-CAD-Model of the primary circuit with shielding walls (front, top, bottom) [6]. The variant shown is based on the SIEMENS/KWU Convoi design.

The NPP Generations 2, 3 and 4 (Convoy) of Siemens/KWU-PWR are expressed in Microshield by diverse representative nuclide vectors (especially differences in the fraction of Co-60), the existence and thickness of shielding, and the material composition of the components.

The distribution of the total nuclide inventory (in the form of contamination) of about 1 – 2 E + 14 Bq to the components of the primary circuit is done firstly based on reverse simulation, starting from known local dose rates, and secondly based on geometrical and physical considerations.

4 Implementation and results

The generic model allows the calculation of the resulting occupational doses generated by definable jobs and tasks. Fig. 3 shows the associated locations for local dose rate measurement (tip of the red pyramids). For each of the four locations, the components within the direct surroundings have been considered as sources (surface contamination), including those behind shielding walls. Simulations were performed considering deposited contamination, specified for each component of an emptied system.

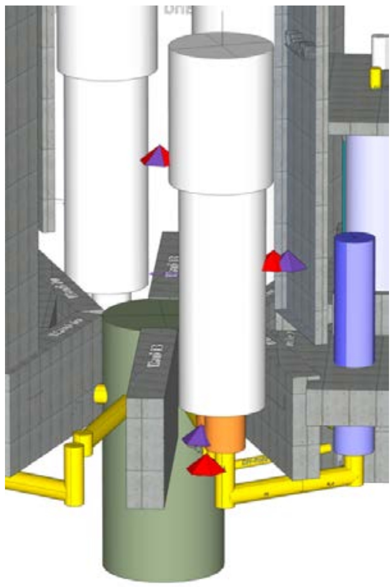

Fig. 2 Specification of representative points in the surroundings of the steam generator during dismantling. The highest local dose rate is generated in the lower region

In the precursor study revision jobs were simulated and benchmarked with measured data from German PWRs. Due to the contemporary planned final shut-down of the three Convoy plants (besides other) until 2022, dismantling work was set into focus of simulation. Simulation was conducted and results compared for Convoy plants and for plants of the older Siemens/KWU-Generations 2 and 3. Furthermore, by comparative simulations the question was answered if full system decontamination in Convoy plants before dismantling leads to a benefit that justifies this measure. The determined dose saving during dismantling works at the steam generators caused by the decontamination is remarkable. An abdication of decontamination at this location would lead to doses much higher than the occupational job dose during steam generator dismantling in a decontaminated Generation 2 facility.

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Session 5. UK Regulatory Approach to ALARA in Light Water Reactors at the Design Stage

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Light water reactors (mainly Pressurised Water Reactors (PWRs)) are intended to replace Advanced Gas Reactors as the main nuclear elements of the UK's planned energy mix.

Experience of civil PWRs in the UK is limited to the Sizewell B unit, however worldwide there are around 280 units in operation. Although progress in reducing occupational exposures in these plants has been mixed, there have been considerable successes by certain operators and in particular families of plants.

The Office for Nuclear Regulation (ONR) is responsible for regulating on site risks and doses to the workforce for operating and proposed new nuclear power plants in the UK. A key part of the ONR approach to ensuring that doses and risks for the UK new build programme are reduced to levels that are ALARA is by ensuring that established relevant good practice in areas such as chemistry control and material selection is applied to new reactor designs.

This presentation will cover the ONR approach to ALARA, particularly the use of relevant good practice and its application to the design of new reactors through the Generic Design Assessment process applied to all new reactor designs proposed for new build in the UK.

Session 5. Design, Modification and Maintenance of Contamination Barriers in Radiological Controlled Areas as a Challenging Part of Radiation Protection in Nuclear Facilities

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The contamination barriers in radiological controlled areas represent the most important protection measures concerning the safe enclosure of radioactivity. Flaws in design and failures as a consequence of deficient maintenance or inadequate consideration of modifications may result in uncontrolled release paths and in prohibited spread of radioactivity. In particular, after final termination of operations as well as during decommissioning the integrity of barriers may be inadvertently compromised.

This contribution describes a systematic approach to contamination barriers in nuclear facilities. The approach takes into account twenty years of experience from inspections and is to be formulated in a forthcoming guideline of the Swiss Federal Nuclear Safety Inspectorate.

Session 5. Ringhals Status of Readiness for Lens Dose Limit Change

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Background:

Swedish regulatory proposal on basic requirements for practices involving ionizing radiation incorporate parts of the provisions covered by the revised EU Directive on basic safety standards for protection against the dangers arising from exposure to ionizing radiation (2013/59 / Euratom). This Directive should also follow new ICRP guidance on the limit for equivalent dose for the lens of the eye in occupational exposure.

Many countries, including Sweden, are planning on implementing a reduction of the lens dose limit to the same value as the Effective Dose limit.

Many aspects of radiation protection to the lens have been identified to be more significant, including field measurements, dosimetry, eye protection, and optimization. The purpose of this presentation is to share with the rest of the industry Ringhals radiation protection program changes and practices to be prepared for the dose limit change.

Session 5. The Brief Introduction of Eye-Lens Radiation Dose Survey in Chinese NPPs

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The results of recent epidemiological studies showed that the threshold dose of cataract formation was decreased to 0.5Gy from 5Gy. Therefore, ICRP and IAEA decreased the equivalent eye-lens dose limit from 150mSv per year to 20mSv per year averaged over 5 consecutive years. This change will trigger some challenges for radiation protection management in nuclear power plant, such as: The methods of lens of eye dose monitoring, measurement program in NPPs, the status of occupational exposure of eye-lens in NPPs and the problem of lens of eye protection and so on. In order to answer these questions, it is very import to understand the plant radiological conditions for eye-lens. Therefore, preliminary eye-lens dose survey projects had been carried out in several Chinese NPPs since 2014. The information of beta spectrum, gamma spectrum and the value of H*(10), H'(3), Hp(10), Hp(3) was collected during the measurement. Then, the ratios of each other were evaluated. The survey locations covered the system of RCS, RRA and RCV and so on. The wokers, who carried out the maintenance work of valves, clearance of pool, were chose to monitor the eye-lens dose. The results showed that about 66.1% of the ratio of Hp(3) to Hp(10) for different people mainly concentrated between 1.0 and 1.7.

Key words: lens of eye dose, dose survey, Hp(3)

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Session 6. Operational radiological protection regulatory treatment applied to the implementation of the Containment Filtered Venting System and the Alternative Emergency Management Center

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In the frame of post-Fukushima actions related to beyond-design accidents, CSN developed a program aimed at protecting plants against other severe events that might be induced by malicious acts and seriously impact the installation safety, the environment and public health.

Actions being requested by CSN are focused on the mitigation of the consequences of these extreme situations. These actions were performed in a separate but in a coordinated process with the actions covered by the stress test and the ENSREG peer reviews. As the result, CSN stablished a regulatory framework (technical instructions) regarding the process and to define an implementation plan consistent with the one already prepared for the incorporation of the stress test conclusions.

In this context several challenges were identified during Plan implementation. This paper is focused on two of them: Alternative Emergency Management Centre (AEMC) and Containment Filtering Venting System (CFVS), at all Spanish nuclear power plants. An overview of the CSN requirements and evaluation process considered for the implementation of these two actions is provided from the operational radiological protection point of view.

Related to CFVS, technological solutions proposed by NPPs owner are shown together with the local manual actions foreseen in the system implementation and the conditions compromising the performance of the manual interventions. Methodology proposed by the licensee and resulting doses rates and intervening personnel effective doses were analyzed.

In relation with AEMC, evaluation process was focused on radiological considerations in the building construction (radiological risk and zones classification, personnel and materials routes) to get a successful supervision of the personnel intervening in the emergency and contaminated materials. Influence of radioactive emissions to the environment and contamination of the ventilation HEPA filters were two of the key issues to define shielding needs.

Finally, paper shows lessons learned based on regulatory framework, NPPs proposals, difficulties faced in the implementation process and the needed compromised solutions along evaluation process.

Session 6. Ringhals Evacuation Centre (Rec) – A Mobile Contamination Scanning Unit For Accident Situations

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Background & Problem:

To keep up with the increasing demands post Fukushima of personal safety for the employees, Ringhals has run a development effort aimed at improving the emergency preparedness of the plant. The effort ran for 5 years and was concluded in 2017.

As part of this, a mobile contamination scanning unit has been developed and practiced several times during realistic conditions. The challenge was to get the unit “up and running” within two hours from the alarm call.

The exercises has been evaluated continuously and REC has further developed to a well-functioning unit that can face current and future demands.

Ringhals wants to share the knowledge that have emerged during the development and implementation of REC and contribute to discussions that can give us an exchange of experiences.

Session 6. Lessons Learned from the Analysis of the Return of Populations following the Lifting of Orders of Evacuation after the Fukushima Accident

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CEPN, France

The conditions for the return of populations to the territories evacuated after the Fukushima accident raise a number of new issues in the context of post-accident management for radiation protection experts. The analysis of the situation reveals the complexity of the situation and the complexity of the individual and collective decision-making processes. It also emphasizes the need to respect the different choices. In this context, radiological considerations are only one dimension of the problem that individuals and communities face.

The lessons to be presented firstly deal with the dynamics of evacuation (modeling, prediction and organization of evacuation) which has strongly influenced the issue of managing the return of populations. Social and especially family dimensions will also be emphasized, particularly related to the organization of temporary housing. Then, lessons will discuss the organization of zoning, the radiological criteria and the effects on health, by analyzing in particular the temporal dynamics during the six years since the accident of March 11, 2011. The dynamics of the return with the problem of the lifting of evacuation orders over time will be addressed by highlighting the difficulties encountered and the specific situations of the various communities. The question of the development of the radiological protection culture and its long-term role will also be discussed. Another key element concerns the effects of the compensation system. Without going into the details of the mechanisms put in place, the questions raised by this system will be presented. Another highlight will concern the multitude of actors involved in the issue of the return of populations with the difficulties of coordination of these different actors. Finally, the question of the future will be evoked with the concerns expressed by several municipalities and the possibility of restoring the attractiveness of the territory for new inhabitants.

Session 7. Performance Monitoring of Radiological Safety Policy at Cernavoda NPP

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Based on industry and internal operating experience CNE Cernavoda developed a comprehensive set of performance indicators for both occupational exposure control and radioactive materials control processes.

Occupational dose indicators have been defined for individual and collective dose of workers. More specifically, average and maximum annual individual internal and total dose of workers, number of unexpected exposure events exceeding the administrative dose limits, number of unplanned exposure events. For indicators related with annual collective dose of workers, we have chosen total annual collective dose per station, the ratio of collective dose during the planned outage to that of power operation, collective dose for each work category, collective dose for high dose jobs, annual collective dose for worker's organization.

A radioactive material control process has been implemented in order to reduce the risk of contamination being inadvertently released outside radiological controlled zones and to maintain internal exposure to a minimum level. Appropriate contamination control practices are in place to improve worker efficiency and, thereby, reduce personnel dose. Performance indicators have been established to control the volume of radioactive waste generated during operation, and maintenance activities, the unexpected contamination events. For the environmental impact and public dose the indicators are the amounts of radioactive effluent in air and water released into the environment from the plant.

Performance indicators have also been an efficient tool for a consistent communication of management expectations regarding radiological safety policy, and for reinforcing the standards of excellence in radiological protection

Session 7. Analysis on Occupational Exposure of Radiation Workers in Korea based on KISOE Database (2007~2016)

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Korea Information System on Occupational Exposure (KISOE) is in operation at Korea Institute of Nuclear Safety (KINS). It is to analyze the occupational exposure of radiation workers in Korea for the purpose of improvement of radiation protection program for radiation workers. It has databases of dose records of occupational exposure for radiation workers in various fields including nuclear energy, industry, research, education and medical application, etc.
Various types of temporal trends on occupational exposure were analyzed by using database of KISOE. The analyses were performed on temporal trends of number of licensees, radiation workers, annual collective dose and annual average dose for detailed categories of license types over a period of recent 10 years (2007 ~ 2016).
Number of radiation workers has increased gradually during the past decade. Nonetheless, annual collective doses have been kept at the same level and annual average doses have continuously gradually decreased for various fields including nuclear power plants. Based on the analyses, it could be concluded that radiation protection programs for radiation workers have been continuously improved in Korea.

Keywords: KISOE, Occupational Exposure, Radiation Protection, Radiation Worker, Korea

Session 7. How to use the ISOE Database?

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The ISOE database includes information for 457 reactor units in 31 countries covering about 85% of the world's operating commercial power reactors. As the world's largest database on occupational exposure from nuclear power plants, the ISOE database provides an important resource for ISOE participants to perform benchmarking analyses and exposure trends at various levels. The database is available to ISOE participants through the ISOE website.

The presentation provides insight into the use of database for optimising radiation protection.

Session 8. Management of Alpha Emitters in the RCS at EDF and Comparison with International Utility Approaches – Presentation of the Data Collected from an ISOE Questionnaire

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The presence of long half-life alpha emitters in the primary circuit is of particular concern because alpha emitters raise problems for maintenance operations, for purification circuit filters and effluent management, and for dismantling. In order to ensure worker safety, EDF, as other utilities, established range of prevention and protection means, based on primary coolant activity, surface activity or dose rate measurements.

The ".ppt" slides present the EDF practices in the field of alpha emitter management and compare its approach to some international ones. The data of the international utilities come mainly from the data collected through a questionnaire sent to the ISOE network members in 2016. Except EDF, the data are provided by Engie Electrabel, Dominion, CEZ, CGN, the NPPs of Loviisa, Cernavoda, Sizewell B, Forsmark and Leibstadt.

The common strategy of EDF and the respondents to the questionnaire to avoid workers exposure to alpha emitters can be described as based on a three step graduated approach:

- Monitoring of fuel integrity during operation,
- Detection and measurement of alpha emitters on the walls of the primary circuits,
- Detection and measurement of alpha emitters on the working area.

Nevertheless, for each step, the parameters and criteria considered by the utilities to implement actions are different.

First, utilities consider various indicators of fuel rod defect: some utilities use the dose equivalent in iodine 131 activity, the isotope activities of xenon, krypton, iodine and cesium or ratio of activities (133Xe/135Xe, 134Cs/137/Cs, 85Kr/133Xe for example)… Several utilities use the WANO Fuel Reliability Indicator (FRI). Most of the utilities use the iodine 131 indicator. Utilities can share a common indicator, but the criteria to trigger an action are often different: Both Engie Electrabel and Sizewell B NPP use the dose equivalent in iodine 131 to detect a fuel rod failure, but the first one established a threshold at 2.34 MBq/kg, whereas the second one has a threshold at 0.2 MBq/kg. Apart from gamma spectrometry in ^{133}Xe, ^{135}Xe, ^{131}I, ^{134}Cs and ^{137}Cs to evaluate the fuel integrity, EDF uses the indicator "iodine 134 activity" to detect a fuel rod failure with fissile material dissemination (with several thresholds). EDF considers that, for a given fuel cycle, there is a fissile material dissemination when the primary activity in iodine 134 increases more strongly than that due to the natural evolution under neutron flux of the fissile material (residual contamination) which is into the RCS at the beginning of the cycle. The alpha-emitters activity is also monitored as a complementary indicator of the iodine 134 activity.

The conditions to perform the next step – surface contamination estimation of alpha-emitters – vary among the utilities: EDF, Engie and CGN perform alpha measurements only if a fissile material dissemination has been identified during the cycle. At EDF, swipes are performed under the reactor vessel head or in the pressurizer manway if the primary activity in iodine 134 is higher than the value of A+1000 MBq/t defined in the radiochemical specifications (where A depends on the ^{134}I activity at the beginning of the cycle, the burn up of the fuel and the previous dissemination in the RCS) or if the primary activity in gross alpha is above 1 Bq/l. Some other utilities, like Dominion and CEZ or the NPP of Forsmark and Leibstadt perform alpha measurements systematically at the beginning of the outage.
Regarding the kind of measurements, their location and the criteria to consider the unit at "alpha risk" they are also differences between utilities. Some utilities proceed to counting, gamma or alpha spectrometry. To identify

the unit at "alpha risk" they use various parameters and criteria (thresholds): surface activity in alpha emitters, ratio bêta-gamma/alpha (as recommended by EPRI guidelines). The location of the measurement or sampling can be under the head vessel, the pressurizer manway, on the reactor vessel, the primary circuit, the secondary circuit... The unit is said at "alpha risk" if the alpha-emitters surface activity is above 8 Bq/cm2 (CGN and EDF) and 20 Bq/cm2 (Engie Electrabel). Dominion and the Cernavoda NPP defined a three level scale regarding the alpha risk of the unit based on the β/α ratios, but the thresholds are different: the low risk level is reached if $\beta/\alpha > 15000$ (Cernavoda) or 30000 (Dominion). The Forsmark NPP established also three levels of alpha risk but the indicators used are alpha activity and β-γ value (from swipes). The first level ("blue") is defined for β-γ < 4 Bq/cm2 and alpha activity < 0.4 Bq/cm2.

If the previous thresholds are reached, the third step is performed.
On the working place, alpha measurements are performed through swipes for most of utilities. Usually, to detect an "alpha risk at the working place" or to identify its level, the same indicators and criteria than those define for the alpha risk of the unit are used. Thus, there is an "alpha risk at the working place" if the alpha activity of the surface is superior to 0.4 Bq/cm for Loviisa NPP, 8 Bq/cm2 for CGN and EDF and 20 Bq/cm2 for Engie Electrabel. Dominion and Cernavoda NPP use the indicator β/α, Forsmark NPP the indicators β-α and alpha activity. The Leibstadt NPP reports an alpha risk on the working area if the alpha emitter concentration exceeds 1/20 of the concentration which leads to 1 LDCA.

Once an alpha risk has been identified on a working place, specific protection and monitoring means are implemented. Depending the utilities, zoning, information, training, collective or personal protection equipment are requested. The dedicated monitoring means encompass swipes, air sampling (Dominion, Cernavoa and Leibstadt), personal dosimeter, lapel personal air sampling (Dominion) and bodily sampling (Engie Electrabel). At EDF, depending the work to be done, the following actions can be implemented: ventilated or non-ventilated anti-contamination tenting, self contained breathing apparatus, leaktight ventilated protective suit (when high risk of contamination dispersion), protective suit, aerosol radiation monitoring (alpha and beta), … In case of alpha risk on the working place, bodily samples are collected at the end of each working period.

Last, to detect an alpha contamination to workers, all utilities implement a set of radioactivity control arrangements: hands, feet and other surfaces control at the exit of the working place, radiation detectors when going through turnstiles to exit the plant (=C1, C2 and C3 portiques), anthropogammametry. As previously mentioned, only Engie Electrabel performs analyses of bodily samples as a routine. The others use such analyses to confirm a suspicion of alpha contamination.

Session 8. Organization to Fight Against Workers Internal Alpha Contamination in Decommissioning Works at Saint-Laurent A

J. Laurent (EDF/DP2D), B. Boussetta (EDF/DIPDE), G. Ranchoux (EDF/DP2D)

Consequently to several alpha contamination events that occurred at Saint-Laurent A Nuclear Plant currently in decommissioning, an ambitious program has been rolled out to fight against alpha contamination and therefore against internal contamination of workers.

This action plan takes into account:

- a better control and survey of the workplace radiological environment,
- a better adequacy between Personal Protective Equipment (PPEs) and the risk assessment,
- a reinforced training and technical clearance to mitigate the human factor effect in normal and emergency situations,
- a reinforced technical survey during the work (dressing/undressing phases especially),
- medical survey requirements (nose-blowing management and analysis, faeces exams, biological analysis time reduction),
- a reinforced assess management in airlocks.

Session 8. Internal Dose Assessments at Forsmark NPP

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The dosimetry unit at Forsmark NPP uses whole body counting (WBC) to check for internal contaminations. For this purpose there is one gamma detector (HPGe) in a chair set-up. Calibration is done on a realistic phantom (Livermore) with different geometries. For conservative reasons lung geometry is mainly used. Internal contamination without gamma-emitting nuclides is very unlikely, and no analysis of biological samples is performed at the site. In the case of need for analysis of biological samples an external laboratory will be used.

Whole body counting is performed on four occasions; a) planned screening of certain personnel (>4 times a year), b) before and after high risk jobs, c) event based, and d) on personal request. An effective internal dose is calculated in case the preliminary dose exceeds 0.25 mSv. In such cases, additional WBCs are performed and IMBA Professional Plus is used for dose calculation.

In recent years two events have led to a calculation of an effective dose of ≥0.25 mSv. One event occurred due to high levels of I-131 in the air, released from reactor basins after the removal of the reactor vessel lid during outage. The second event occurred in a work-shop and involved mainly Co-60 in aerosol form. This paper will share the conclusions for these two events and the general methodology for internal dose assessments at Forsmark.

Session 9. New beta probe for contamination monitor under a gamma background

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Since several years, CEA-LIST has been working on a new approach to discriminate beta contamination in a high gamma background. This discrimination is based on the coupling of a rapid decay time scintillator with a slow decay time scintillator which can be also named dual plastic phoswich scintillator or $\Delta E/E$ discrimination technics. In this framework, several topics as material, algorithms and integrated system have been deeply studied [1,2].

Recently, Nuvia company and CEA-LIST have jointly launched a research program to develop an innovative solution based on a new detector integrating the capacity to measure a beta contamination as low as 0.4 Bq/cm² in a high and fluctuating gamma background level of approximately 40µSv/h in 4 sec. of measurement. Several test campaigns carried out at CEA Saclay and NUVIA have proved that the new measurement system is insensitive to gamma background (intensity and energy) and also insensitive to the probe position relative to gamma background.

This paper presents in detail several results and the performance of the developed system which is the basis of the new NUVIAtech brandname contamination monitor.

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Session 9. ALARA & Management of internal exposures at CNE Cernavoda

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Energy production in nuclear facilities embraces a wide range of human activities, which are associated with a potential risk of internal contamination by radionuclides.

Despite a careful planning for radioactive source control, the release of radionuclides in working environment could lead, infrequently, to radioactive material intake by workers. CNE Cernavoda developed a comprehensive dosimetry program, including internal exposures.

During normal operation of a CANDU nuclear power plant significant tritium quantities are generated. Through design solutions that have been implemented and continuous monitoring we manage to control the tritium losses from the reactor systems and keep them as low as achievable.

In case of a CANDU type reactor tritiated heavy water (DTO) is the major contributor to the internal dose of professionally exposed workers – weighting up to 30% of the total effective dose. Internal contamination with other radionuclides than tritium is infrequent, usually having a negligible contribution to occupational exposure.

CNE Cernavoda developed a special strategy in order to control workers' internal exposures to tritium and dedicated programs are running to implement this strategy: improvement of radiation protection practices, increasing equipment performances, leakages prevention through maintenance program, air moisture control and finalize the de-tritiation facility.

Special dryers are designed and are used to remove moisture from different ventilation systems of a CANDU reactor in order to maintain tritium in air concentration and gaseous tritium emissions below the limits established by the national authorities. Vapor Recovery Systems are designed to control tritium in air concentration and to recover heavy water loss from PHT and Moderator Systems and to control the air circulation, providing atmosphere separation between different areas of the Reactor Building.

Various respiratory protection equipment are used by workers performing radioactive contamination risk work, as required by Radiation Work Permit.
Internal contamination of workers may occur due to intakes of radionuclides as a result of several activities, mainly during undressing or when the airborne contamination is unexpected. The most common route of entry of a radioactive contamination into the body of a worker is by inhalation of airborne radioactivity. A strict control of contamination events allow to promptly identify gamma radionuclide intakes and initiating bioassay procedures for internal dose calculation.

This paper presents the actual situation of workers internal exposures and emphasize the results of the ALARA policy promoted by Cernavoda NPP management for reduce internal exposure.

Session 10. A New Application for Industrial Radiography

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The industrial radiography activity is known to be submitted to strong constraints (personal safety, planning). EDF's Grand Carenage program aims at extending the life of Nuclear Power Plants beyond 40 years. It increases the maintenance activities and therefore the volume of radiography activities with acute planning constraints.

Radiography activity requires to demarcate the boundary of a controlled area, within which only radiologists are admitted during radiographic tests. Therefore, preparation requires a good spatial understanding of the operation area, and especially of all the accesses (including ladders, stairs, elevators…). The Reactor Building has a complex spatial structure and is not accessible during the preparation phase, making this preparation particularly difficult.

EDF R&D has developed the application "Tir Radio". This application is based on a complete digitalization of the reactor building, currently in progress at EDF on some representative reactor buildings. A virtual model is made from thousands of laser scans and panoramic photos, allowing to describe very precisely the facilities, their accesses. It contains algorithms capable of analyzing this virtual model to find every possible paths in the reactor building.

The application has a simple user interface and helps radiologists to prepare their activities at their desks by:

- presenting a clear view of the building to simplify the definition of the restricted area
- automatically finding all paths leading to the radioactive sources to easily detect the unprotected accesses to the restricted area
- facilitating communication to better coordinate the activities and to secure the planning of maintenance operations
- and if necessary, quickly updating the design of the restricted area

Session 11. Spent Fuel Storage Rack Replacement

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The nominal capacity of the Spent Fuel Storage Rack (SFSR) in the Spent Fuel Pool (SFP) of each 1300 MWe unit is around 630 cells. Each cell can contain one fresh or burnt fuel assembly.

The under-criticality of the spent fuel storage is guaranteed by a neutron-absorbing material, “BORAL”. These boral plates, by an under-water corrosion phenomenon, swell over time to locally form blisters inside the cells. Owing to this swelling up observed in a few 1300 MWe units, some cells become unavailable, which create less spent fuel storage rack capacity and can potentially generate, eventually, operation and safety issues.

1300 MWe units are concerned in various ways by the swelling issue. The most critical situation in France is for Penly 2, Cattenom 3 and Nogent 2 units. The Golfech 1 & 2 and Belleville 2 units progress with a slower kinetic. The remaining 1300 MWe units are mildly affected (a few cells unavailable without progression). The spent fuel storage rack of Penly 1 was already replaced in 1998.

In 2009, EDF decided to authorize the SFSR replacement for the six most affected units before the unavailable cells generate operation and safety issues.

In 2016, eight of ten racks had been replaced in Cattenom 3. The Nogent 2 and Penly 2 operations are respectively scheduled for 2019 and 2020.

During the spent fuel storage rack replacement operations, the racks are decontaminated with adapted tools. At the end of decontamination dose rate measurements are carried out to make sure that the racks:

- Have a residual contamination level sufficiently low to continue the operation without radiation risk for the staff on-site;
- Have a dose rate compatible with transport regulations when the rack is inside the container;
- Have a residual activity compliant with the treatment rack as a waste.

To assess the rack activity, a MCNP modelling has been set up and validated with the feedback measurements from Cattenom 3. Activity criteria under water have been defined to respect the dose rates limits in air. This modelling allows to forecast the overall activity of the old rack depending on the dose rates measurements of the different sources: cell bottom, vertical wall and other parts.

Session 11. Operational Experience of the First Dry Fuel Storage Campaign at Sizewell B NPP

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This presentation describes operational experience gained during the first campaign of loading spent irradiated fuel into dry casks for long-term on-site storage.

Historically the UK nuclear fleet has consigned its spent fuel for reprocessing. However since commissioning Sizewell B, the sole PWR in the UK nuclear fleet, has stored its spent fuel on-site in the Fuel Storage Ponds. After more than twenty years of operation the maximum capacity of the Fuel Building was being approached. Therefore, after due consideration of nuclear safety, environment and reliability, a decision was taken to implement a dry fuel storage system using the Holtec dry cask system.

This presentation describes the radiological protection implications of the first campaign of loading irradiated fuel into dry casks and placing them into their long-term storage location. The presentation further describes techniques used to characterise and manage the radiation source term including the neutron dose component. The campaign of seven casks saw a progressive reduction in doses per cask as operating experience was used to drive process improvements. The presentation will describe the major radiological protection improvements that influenced the dose trends and the lessons learned for the next campaign, expected in 2019 or 2020.

POSTERS

Poster 1. 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 controlled areas of nuclear sites in EVEREST[2] mode. This research falls within the plan of strengthening the prevention of risks arising from radiological (contamination, ...) or conventional (heat stress, ...) hazards. But for the first time ergonomics and comfort are recognised as key aspects needing addressing and thus thoroughly investigated by in-depth studies in both technical and human fields.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.

To date, the studies have focused mainly on protective suits. First results on ergonomics and comfort as ways to lessen the risk of contamination have been implemented in early prototypes, which were tested at the training platforms of different nuclear power plants. The feedback from these essays will allow manufacture of initial pre-series of protective suits which will be tried out on a larger scale in real work sites in controlled areas.

Further perspectives are envisaged and aim to integrate into the workwears passive technical innovations to improve the working conditions of workers vis-à-vis of the risk of contamination and heat risk.

Finally, this contribution will present the advances made in the design of the protective suits and how the prevention of radiological risks will benefit from their ameliorations on ergonomics and comfort.

[2] EVEREST Project consists to enter in 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.

Poster 2. Study of the feasibility of a detector dedicated to the discrimination β/γ

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The contamination monitor is essential for nuclear operators to assess surface residual activity such as beta activity for radiation protection purpose. To this aim, EDF R&D has proposed a research project in order to study the feasibility of an innovative contamination monitor to detect beta surface activity in situ which is until now performed ex situ. The main objective is to detect 0.4 Bq/cm2 beta surface contamination in a high gamma background up to 100 µSv/h. This demands the detector to have a high beta/gamma discrimination performance. Two technologies have been investigated: one based on a Phoswich detector and the other one based on a gas detector. Geant4 simulations have been carried out to optimise the configuration of the chosen detector such as geometry, materials, etc. To validate the simulations, a simulation-optimised prototype has been fabricated and tried experimentally by means of beta and gamma sources in a laboratory framework (LPC Caen). Furthermore, an experimental campaign has been conducted in a nuclear power plant (nuclear power plant of Chinon) in order to evaluate the performance of the developed prototype in a realistic gamma background environment. The Geant4 simulations and the experimental work will be discussed in this presentation.

Key words: contamination monitor, beta surface activity, beta/gamma discrimination, Geant4 simulation, Phoswich detector, gas detector

Poster 3. Simulation and Prediction of Dose Rates at Interim Storage for Spent Nuclear Fuel at Dukovany Nuclear Power Plant

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The objective was to create the model of interim storage of spent nuclear fuel situated at the territory of Dukovany nuclear power plant to simulate and predict dosimetric characteristics, i.e. ratio of photon/neutron dose rate. The interim storage covers a wide range of stored spent fuel loaded into CASTOR®440/84 casks. Due to the ALARA optimization the model was designed to predict large changes of dosimetric characteristic in interim storage. Model was created by SCALE 6.2. (ARP-ORIGEN, MAVRIC) and was widely validated using available data of mixed field components at the interim storage and stored casks, providing accurate and easy to use tool for obtaining dosimetric characteristics in interim storage.

Keywords: interim storage, mixed fields, SCALE, CASTOR®440/84

Poster 4. A Portable Quantitative High-Resolution Gamma System for Waste, D&D, and Emergency Measurements

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The Canberra ISOCS system has been a highly successful and technologically innovative product since its introduction about 20 years ago. The combination of field-portable HPGe detectors, flexible shield/collimator set, and accurate mathematical efficiency calibration software made in-situ gamma spectroscopy measurements very practical. Good quality in-situ measurements are economically beneficial large and complex item assay in comparison to the cost and time of extracting a sample for laboratory measurements. And where the item is not homogeneous [most of the time] an assay of a large part of the item has been shown to be more accurate than a few widely dispersed extracted samples [the normal case]. However, the size and weight and cost and complexity of the full-size ISOCS system limit its usefulness in some situations. A companion device was recently introduced. The gamma sensor is a 1 cm3 CZT detector. It is surrounded by a tungsten shield/collimator set with a tripod for support. The detector is powered solely by a USB connection from a laptop PC. The full power of the ISOCS efficiency calibration software can be used with this detector. This entire package is less than 15kg, needs no cooling, and is quite rugged, in comparison to the 120kg for ISOCS with sensitive HPGe detectors which need LN cooling. And this CZT package is about 1/3 the cost. This is a very complementary device for the many current ISOCS users, especially for quick initial evaluations or simple situations where the full power and sensitivity of the HPGe ISOCS are not needed. Operating nuclear power plants or those in the initial phases of decommissioning are prime examples. Their complex layout often precludes the deployment of the full-sized ISOCS system, whereas the CZT system can be hand-carried there and very quickly setup for a measurement. The small detector size, and therefore low efficiency, is not a disadvantage, and in many cases is an advantage due to the elevated radioactivity of the items. This can be used to assay items in tanks, spills on floors, activity on filters, and waste in drums. In the event of an emergency, it can be quickly deployed to measure gas or liquid effluent streams, air particulate and iodine filters in the field, or even in-vivo thyroid activity. Examples of measurements at nuclear facilities are shown, and Minimum Detection Limits are presented.

Poster 5. A Device for Continuous Repeating Real-time Assay of HPGe, CZT, or Scintillation Gamma Spectra

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There are many situations where a continuous repeating series of quantitative gamma spectral assays is desirable. These include measurements of fluid going through a pipe, of items on a conveyor passing under a detector, a detector aimed at the ground on a moving platform, or a detector in free air as a plume of radioactivity drifts by. While Canberra and many others have built and deployed many of these devices, they have all been custom built for that particular application, and rather expensive. The Data Aggregator [DA] is a general purpose device that is designed to fulfill many of these applications without modification, but can be fairly-easily modified if needed. The DA is a small [12x18x5cm], low-power [3W] box that accepts the input from external MCAs, controls the acquisition cycle, analyzes the results, and transmits them to the outside world when needed. The detector-MCAs supported are HPGe with Lynx MCA, NaI or CeBr or LaBr with Osprey MCA, and CZT or CsI with internal MCAs. The DA internal software used for analysis is the full standard Genie package. This is very widely used and well accepted in the industry. When the spectral files are exported, the analyses can be reviewed and confirmed by Experts, and reanalyzed if needed. The DA hardware is totally autonomous. As soon as power is applied, it starts the pre-programmed sequences of acquisition and analysis. Multiple simultaneous analysis sequences [called Workflows] can be in operation simultaneously, e.g. with different count times and nuclide libraries. Only the detector, MCA, and DA is required in the field. A PC can be connected to the DA via WiFi, USB, or Ethernet. The PC is used to setup the DA, and to readout the results. The DA also supports [with application-specific programming] external Inputs/Outputs either for recording with the analysis results or for use in the computation of the result, e.g. temperature, flow rate, pressure, …]. The DA also has an internal GPS receiver and external antenna. Previous applications of the DA that will be discussed are stack gas activity where samples were extracted and measured in a shield with a HPGe detector and Lynx MCA, and reactor coolant activity where the shielded CZT detector and MCA was aimed at various sections of reactor primary coolant during a maintenance outage.

Poster 6. Blayais NPP (France) Drain Pipe Event

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In 2015, the inspection of the drain pipes of BLAYAIS 2 Steam Generators (SGs) showed the presence of important deformations in their extremity near the welded zone, on the Primary Channel Head (PCH).

These deformations were noticed on two SGs: SG1 hot leg side (HL) and SG2 cold leg side (CL). Since the deformation is more significant on SG2, the priority was given to detailed study.

In 2017, SGs were subjected to further inspection and the measurements carried out on SG1 showed an evolution of the deformation compared to those noticed in 2015.

Between 2015 and 2017, there were three operating cycles that is to say three transients of startup-shutdown however SGs were only opened in 2015 (end of the 1st cycle) and in 2017 (end of the 3rd cycle). Consequently much that what was observed in 2017 is the result of what happened in all three cycles.

If we call the deformation « blister », for SG1, the 6mm high blister observed in 2015 had reached 10mm in 2017, two cycles later.

An intervention is scheduled for 2018 on the two SGs to repair the drain pipes. This intervention includes the treatment of the deformations, the insertion of a new drain pipe and the welding of the two extremities inside the PCH.

Considering the high stakes in Radiation Protection with a collective dose approximately 60 man.mSv per SG, a dose rate around 30 mSv/hour and the entrance of personnel inside the PCH for tooling and welding processes, an ALARA approach in charge of optimizing (reducing) the overall exposure has been applied :

- A PCH mock-up has been designed to train the employees and thus optimize the intervention durations
- A specific tool has been designed to avoid the presence of personnel inside the PCH for the treatment of the deformations
- A collaborative Working Group ALARA (EDF Study & EDF Power Plant & Subcontractors) has been set up in preparation phase to define practical ways to reduce the collective and individual doses
- Decontamination processes and specific biological shielding have been studied
- Personal protection equipment has been optimized to facilitate the intervention, ensure the control of the technical gestures with a high level of protection against radiological risks
- The follow-up methods for collective and individual doses have been reinforced to control and anticipate any dose deviation from expected levels

Poster 7. "Analysis of Surface Contamination of the premises of the controlled area and equipment of the primary circuit of the Armenian NPP"

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It is obvious that the contamination of the surfaces of the premises of the controlled area of the nuclear power plant and the equipment of the primary circuit is one of the main factors in the formation of the dose of external exposure of personnel during the reactor planned outage. Such information makes possible of more accurately planning the dose of external exposure of personnel during work in the controlled area.
In addition, such information will be extremely useful in planning the volume of radioactive waste resulting from the decommissioning of nuclear power plants.

The article presents the results of the investigation of the isotopic composition of surface contamination of the premises of the controlled area. Preliminary information has been obtained on the depth of penetration of radioactive substances into various structural and protective materials in the most contaminated areas of the controlled area (floors, walls). It is determined that the depth of penetration of radioactive substances into various coatings does not exceed 8-10 mm.
The work on the investigation of radioactive contamination of the internal surfaces of equipment and pipelines of the primary circuit was carried out. Preliminary data were received and processed.
The cartograms of dose rate values were prepared at different elevations of the reactor compartment of Units 1 and 2, as well as a special building.
The analysis of the obtained results is carried out.

Sampling for determining surface contamination of premises was carried out using wet smears. Samples to study the depth of penetration of radionuclides into various coatings were selected with the help of cores. Samples were measured using a gamma spectrometer with a semiconductor Ge detector and GENIE software support.

As a result, a database was created at the Armenian NPP, characterizing:

- dose rate in the reactor department premises,
- surface contamination of the surfaces of the premises of the strict regime zone and the degree of penetration of radionuclides into various floor and wall coverings,
- activity of deposits on the internal surfaces of the equipment in the primary circuit.

The obtained data will be used to plan the dose loads of personnel.

Poster 8. NuVISION: a portable spectrometric imaging device for searching and identifying radioactive sources

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NUVIA (subsidiary of VINCI group) has developed with the CEA LETI a compact spectrometric gamma camera based on CZT semiconducting detectors, coded aperture and Compton imaging, named NuVISION. The aim is to provide end-users with a portable and sensitive system that allows them to not only detect but also localize radioactive sources and identify radioisotopes, thanks to excellent spectrometric performance over a wide energy range (20-1400keV) and up to 100mSv/h dose rates.
The camera weights 3 kg including battery. Detector energy resolution is 2.5% at 122 keV and 1.5 % at 662 keV. Each gamma event is localized on a 128 × 128 pixel array. The resulting spectral image is reconstructed in real time to identify isotopes and localize activity. The angular resolution is 3 degrees for a field of view of 45 degrees using the coded aperture and 15° for a 360 degree field of view using the Compton imaging. The system is sensitive enough to localize a 50 nSv/h Co-57 source in natural background in less than 1 second and a 50 nSv/h Cs-137 source in less than 1 minute.

Example of image obtained from the camera in a 1 s frame - 3.7 MBq Am-241 source in red and 1.7 MBq Co-57 in green.

Complying with ALARA guidelines drives the emergence of new technologies and processes to minimize dose intake for the personnel of nuclear sites. Quickly assessing hot spots during a nuclear plant outage, or characterizing waste during standard operation or D&D work also became a key driver to deliver projects on time and within budget. NuVISION is a strong asset to assist staff working in nuclear environment such as:

- Radioprotection Management
- Source-Term
- Radioprotection at the Decommissioning Stage
- Radioprotection and Waste Management
- Safeguards
- Emergency response

Poster 9. A high sensitivity Compton Suppressor System for nuclear waste characterization

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The precise characterization of radioactive waste is required in order to optimize its classification management; including transport, inter-mediate storage, and final disposal. For this purpose, non-destructive assays are applied such as gamma-ray spectrometry which is accessible with high-resolution spectrometers, typically high-purity germanium (HPGe) detectors [1]. However, this technique suffers from a large continuum background in the low-energy region, potentially covering smaller photo-peaks of interest in isotopic identification. The measurement of an ^{241}Am activity by gamma-ray spectrometry is, for instance, made difficult by the low energy of its main gamma ray signature (59.5 keV).

This CEA research project investigates a non-destructive assay and deployment of a prototype to identify and quantify radioisotopes in nuclear waste applications from Nuclear Power Plants. The concept integrates an active shielding system based on Anti-Compton principle.

First, a Monte Carlo simulation (MCNP6) was performed in order to simulate the geometry of the Compton Suppressor, which includes a HPGe detector surrounded by a 1.5-inch BGO detector ($Bi_4Ge_3O_{12}$). Then, another program was developed to analyze the PTRAC files generated by MCNP6 [2]. The coincident events from both detectors will not be counted, and the background due to Compton-scattering in the central spectrometer will be reduced.

The simulation results of the anti-Compton are shown with a ^{137}Cs source in figure 1. The upper spectrum is without anti-Compton suppression and the lower spectrum is obtained by applying anti-Compton suppression. As can be seen in the figure, a significant reduction of the Compton continuum is theoretically achieved. Indeed, the reduction factor of background ranges from 2 to 10, depending on the region of the spectrum.

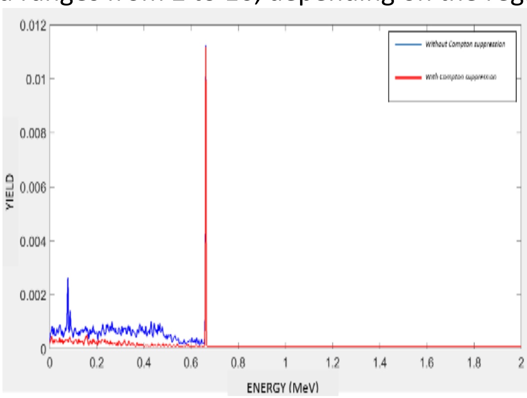

Figure 1: Compton reduction obtained with anti-coincidence treatment of PTRAC files generated by MCNP6

The experimental study of this concept is currently being carried out in our laboratory. The test bench consists in a HPGe detector (2keV at 1332keV) commercialized by CANBERRA France, and a BGO detector manufactured by Crismatec. The first experimental results have been found in promising agreement with theoretical expectations. The present study confirms the cogency of this concept, which will provide a practical means for background reduction relating to nuclear waste characterization.

[1] F. Carrel et al., Characterization of Old Nuclear Waste Packages Coupling Photon Activation Analysis and Complementary Non-Destructive Techniques, *IEEE Trans. Nucl. Sci.*, Vol 61, No. 4, 2014.

[2] R. Coulon et al., Simulation of scintillation signal as a help phoswich systems conception, *NSS Conf*, N-03-144, 2017.

Poster 10. WasteApp - A Mobile Application for a Supervised Process for Radioactive Waste Management

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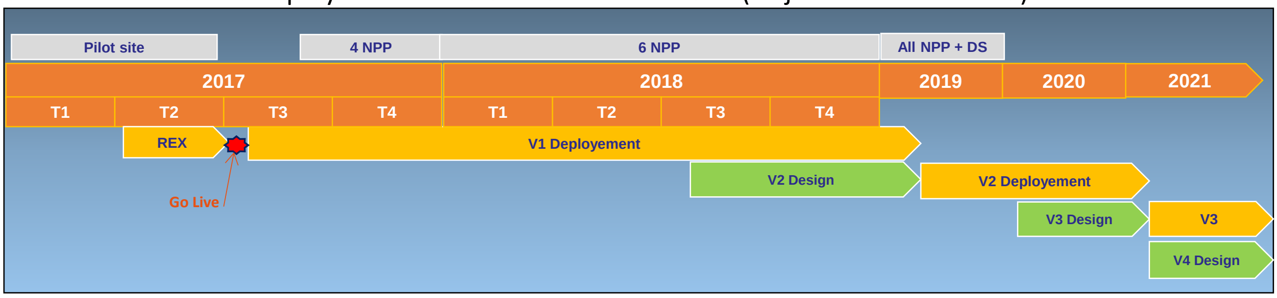
WasteApp

This application makes possible to modernize and simplify the management of radioactive waste in nuclear installations. Thanks to the digital transition, all the parameters are computerized. All the history is traced, and new informations are capitalized along the entire waste packaging chain, then, from the waste package to the disposal site.
The developed tool has a part "mobility" and another "Web". Data is managed in real time.
This application is very innovative for this area of activity; it allows to:

- simplify and modernize the work of operators, - reduce dosimetry, - improve the traceability of waste, and waste packages, - control the regulatory compliance of facilities, - manage stock, - supervise in real time the production,	- anticipate the human needs, material of the activity, - capitalize the production of waste by site in the facility (establish a REX site production), - save time on the entire waste business, - upgrade the profession of waste technicians

Context elements:

Following the Pilot site feedback, the application has been operational since mid-2017 and has been the subject of a validation of the deployment to the entire EDF NPP fleet. (Objective: end of 2019)



The application is evolutionary. Future versions will allow to integrate other needs of the NPPs fleet, the Decommissioning Department, but also the needs of the "waste" business teams of all our units.

REX elements (pilot site, 10 monts):

17,000 scanned bags
(of which 8500 packed in drums)

1200 waste packages created

730 waste packages shipped

Example of gains:

RP waste management => Dose Rate gains for counting packages and containers = 2,9 mSv/Y
Waste management => Time savings for the team = 2500 Hrs/Y

Poster 11. Outage Activity Transport Monitoring at Cernavoda NPP

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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.

Dose rate and gamma spectra surveys have been performed on the reactor faces, vertical feeder and moderator heat exchanger of Cernavoda Unit 1 and Unit 2 starting with 2010.

For both Units, significant differences were observed between "A" and "C" reactor faces, due to Co-60 and Nb/Zr-95. The radionuclides contributors to the fields were Co-60, Zr/Nb-95, Sb-124, and Fe-59.

The radiation field across the reactor faces was affected by hot spots and the overhead sources. The analyses suggest that in order to effectively decrease the radiation field near the reactor face the shielding has to be installed in the space between the end fittings.

The project of gamma fields characterization at Cernavoda NPP will continue with measurement of PHT spent mechanical filters in order to determine the efficiency of PHT purification system and to establish the appropriate actions to increase the purification rates (modification of purification flow, downsizing the filtration pore).

Poster 12. Strontec, a new and fast method for reliable Sr-90 determination

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Strontium 90 (Sr-90) is a well-known radioactive isotope and produced by fission in a nuclear reactor or during explosion of nuclear weapons. It also represents one of the most hazardous radioactive isotopes since it is difficult to detect, both its daughter nucleus ^{90}Y and ^{90}Sr itself emit only beta particles and no gamma rays, and has similar chemical properties to calcium. Thus, it easily accumulates in bone tissues, and due to a half-life of 28.8 years and a rather large beta decay endpoint energy of the daughter nucleus Y-90 of 2.28 MeV can cause considerable damage to the surrounding tissue and increase the risk for cancer.

In the framework of the research project ''New and fast method for reliable Sr-90 determination'' a method was developed successfully to detect Strontium 90 (Sr-90) activity in material samples directly without chemical treatment. The scope of the project was to achieve sufficiently low sensitivity limits and minimum detectable activities (MDA) of Sr-90 (less than 10 % of the release levels of Sr-90 according to table 1, appendix III of Strahlenschutzverordnung;StrlSchV). The measurement was performed with an alpha - beta-low-level-proportional counter at a low background-activity environment on a nuclear power plant site. To determine characteristic limits (decision threshold, detection limit and limits of the coverage interval), a modified general model of evaluation according to DIN ISO 11929 was used.

The developed method is a convenient and quick alternative to radiochemical nuclide analyses hitherto employed in the nuclear field. Especially in the decommissioning process it will be essential to monitor Sr-90 activity in order to release material according to §29 StrlSchV. The minimal effort in time and effort to carry out the measurements and the availability of proportional counters at nuclear power plant sites should allow a straightforward implementation of the newly developed technique.

Poster 13. Process of Implementation of New or Modified RP Regulations

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New or modified regulations, Act and Ordinance with respect to radiation protection will be enforced in Sweden in the near future, beginning in April 2018. The Swedish nuclear facilities have been included in the review process of these regulations but the final versions have not yet been presented.

Changes to regulations are made on a regular basis but the changes we are facing are extensive and far reaching affecting the whole radiation protection organisation.

Oskarshamn NPP will begin by performing a process with gap analysis to identify the differences between the current and the upcoming directives. From these results a plan will be formulated to meet these new requirements by updating the safety analysis reports, processes, existing and new instructions and how to implement them. This presentation will address the road map how to comply with the wide ranging requirements.

Poster 14. Nuclear fuel assembly identification during core unloading/loading using a dual gamma spectroscopy system

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***Abstract* – Core unloading and loading during outage of a nuclear reactor is a critical task and mistakes could degenerate into serious accidents as occurred the 2nd of April 2001 at the Dampierre nuclear power plant (France).**

An in-situ measurement apparatus based on two gamma spectroscopy systems is proposed by the CEA LIST to address this issue. These techniques are the gamma spectrometry using CdZnTe or LaB3:Ce detector and the densitometry measurement.

***Index Terms* — unloading; densitometry; nuclear fuel assembly; burnup; accident.**

I. Introduction

Nuclear fuel assemblies are transported, from the core to the storage pool during core unloading, and from the storage pool to the core during loading. If a fuel assembly is placed in a wrong location into the core, the neutronic configuration will not be the one expected and neutron criticality accident could occur.

Systems from previous works are based on several nuclear measurements such as passive and active neutron measurements, and gamma spectrometry analysis [1]. Moreover, additional ultrasonic inspection and visual examination using a video camera can also be implemented for such purposes [2].

A measurement scheme is introduced in this paper coupling gamma-rays spectrometry and densitometry.

II. Description

The system is immerged into the water of the pool in the path of assembly transit between core and storage pool. The gamma spectrometer (*cf.* Fig. 1) can be medium resolution detector such as CdZnTe diode or LaBr3:Ce scintillator. It is surrounded by a lead collimator allowing to measure signals $S_{^{134}Cs}$ and $S_{^{137}Cs}$ from respectively ^{134}Cs and ^{137}Cs as a function of the height z of the fissile column. A figure of merit I_1 linked to the fuel burn-up is obtained such as:

$$I_1 = \frac{\int_z S_{^{134}Cs}(z)\,dz}{\int_z S_{^{137}Cs}(z)\,dz} \qquad (1)$$

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The signal of a short-life isotope as for instance the ^{131}I is also extracted during the unloading and during the reloading in order to provide an estimation of the cooling time $\hat{\tau}_c$ and compare it with the expected one τ_c.

$$\hat{\tau}_c = \frac{\ln\left(\int_z S_{^{131}I}(z)\,dz\right) - \ln\left(\int_z S_{^{131}I}(z)\,dz\right)}{\lambda_{^{131}I}} \qquad (2)$$

Where $\lambda_{^{131}I}$ is the decay constant of the ^{131}I.

Therefore, the gamma spectroscopy system offers two independents indicators characterizing a given assembly. A test source or a temperature sensor can be embedded into the collimator for detector gain stabilization.

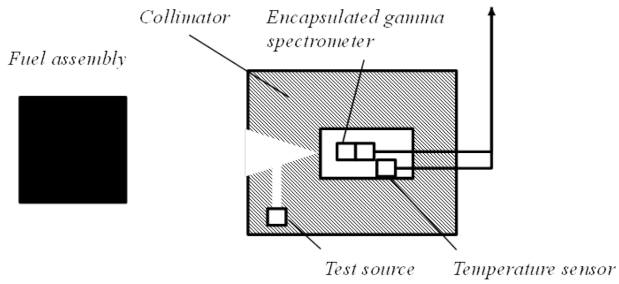

Fig. 1. Schematic view of gamma spectrometry system.

These indicators are completed by a third one given by another gamma spectrometry system placed at the corner of an assembly in order to focus on a fuel pin as illustrated in Fig. 2. It uses roughly the same set-up but a gamma-ray source is set at the opposite of the detector. A density indicator I_2 is calculated such as:

$$I_2 = \int_z \frac{S'_Y(z)}{S'_{Y,0}}\,dz \qquad (3)$$

Where $S'_{Y,0}$ is the signal measure in the absence of fuel assembly and $S'_Y(z)$, the signal measured during the passing of the fuel assembly.

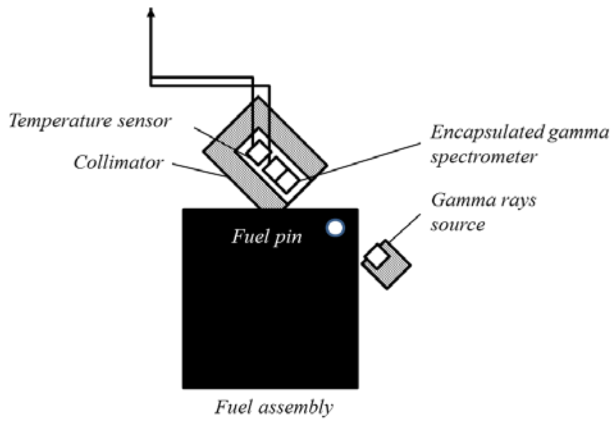

Fig. 2. Schematic view of densitometry system.

Finally the dual system delivers three figures of merit:

- I_1 linked with the fuel burnup;
- I_2 linked with the fuel density;
- $\hat{t}_c$ linked with the storage time of the fuel assembly.

These indicators, physically independent, are able to generate reliable alarms helping operators to make decision about the loading management.

III. Test of the Densitometry measurement

The densitometry measurement has been simulated thanks to a MCNPX model of fuel assembly, source and detector (*cf*. Fig. 3). Density is known to be changed with fuel burnup [3] varying between 10.97 g.cm^{-3} to 10.10 g.cm^{-3} at 60000 MWd.tU^{-1}. A source of ^{152}Eu and a 100 cm^3 CdZnTe diode is considered.

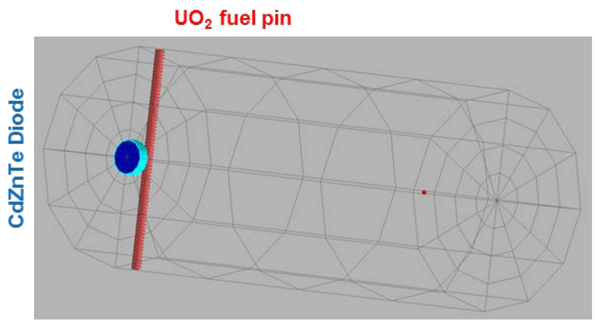

Fig. 3. Schematic view of densitometry system.

Fig. 4 presents the evolution of the figure of merit I_2 as a function of the gamma-ray energy for the two borders values of density. The contrast is maximized at 345 keV where I_2 shows a relative decrease equal to 3.14 % for a decrease of density equal to 0.1 g.cm^{-3} (7000 MWd.tU^{-1}).

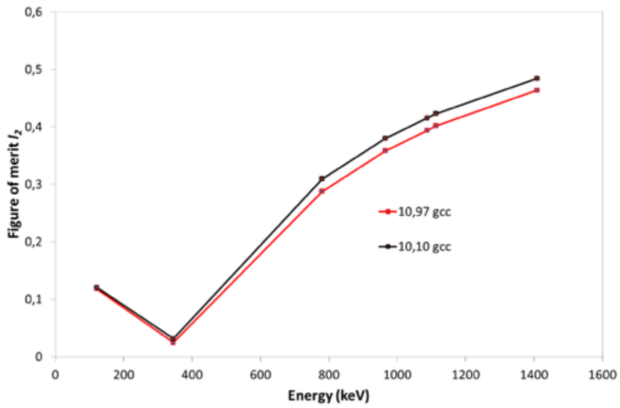

Fig. 4. Figure of merit I_2 as a function of the energy for two border values of fuel density.

The gamma spectroscopy analysis can be optimized by using innovative deconvolution techniques such as the SINBAD code [4], and the level of verification can be improved by the comparison of axial profiles $I_1(z)$ and $I_2(z)$.

IV. Conclusion

A dual gamma spectrometry system is investigated by CEA for nuclear fuel assembly identification to prevent loading accidents which could lead to a criticality accident. The system allows us to provide three independent and redundant figures of merit to verify that the assembly transported from the storage pool to the core of the reactor. The feasibility of the densitometry analysis has been proven by Monte-Carlo simulation and shows promising results.

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