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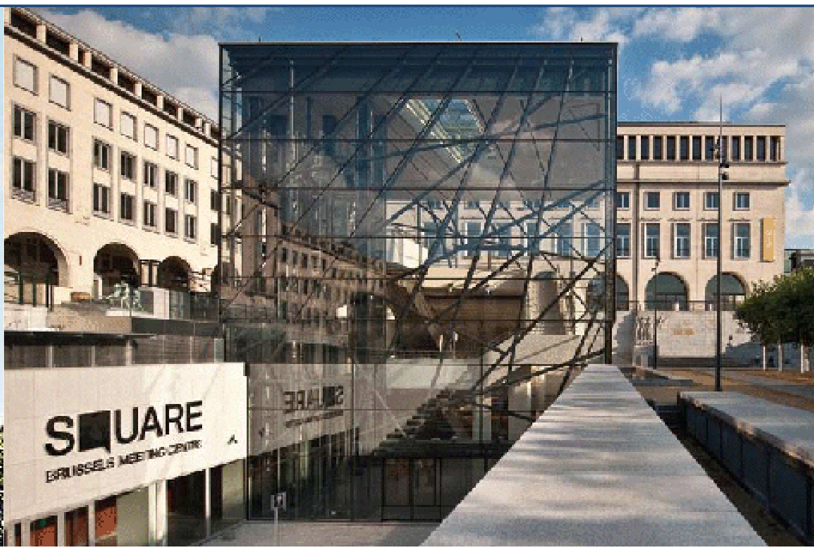

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BOOK OF ABSTRACTS

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POSTER PRESENTATIONS

Session 1. Use of ISOE for the Assessment of the RP Practices and Results at ENGIE Electrabel

Benoit LANCE

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ENGIE Electrabel is licensee and operator of 7 reactors located in Belgium and distributed on two sites, Doel and Tihange. Since December 1st, 2014, the Group ENGIE enforced the development of the Independent Nuclear Safety Oversight (INSO) in all its companies, being inspired by the developed practice at ENGIE Electrabel. INSO consists in providing the capability of the senior management to commission independent assessment of the performance of the organisation, regarding nuclear safety (Radiation Protection (RP) is included in this definition) and receive independent advice where this is appropriate.

The paper is aimed at illustrating how ENGIE Electrabel takes part to ISOE, in order to assess the RP practices and results against the international best practices. The participation to ISOE therefore supports the INSO process in the field of radiation protection.

Several examples will be provided, using the ISOE data and ISOE RP forum. Amongst those, one can already mention:

- The use of the ISOE exposure data, in order to assess the RP results over the period 2005 – 2013 for the nuclear power plants of Doel and Tihange. This assessment was achieved within the framework of the 10-year Periodic Safety Review (PSR), in agreement with the IAEA specific safety guide SSG-25;
- The use of the RP forum, in order to assess the practice of the respiratory protections, within the framework of a specific INSO review;
- The use of the RP forum, in order to benchmark the alarm thresholds set for the emergency electronic personal dosimeters used at Doel and Tihange, against other ISOE operators. Such an assessment was kicked off within the framework of the action plan set up in the aftermath of the Fukushima accident.

The paper will end up with the perspectives of further use of the ISOE information and network, expected to support the INSO process inside the ENGIE Electrabel company.

Session 1. New radiological control of areas in Fukushima Daiichi NPS

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Brief introduction of the presentation

Fukushima Daiichi Nuclear Power Station (1F) was hit by the big earthquake and tsunami, which caused the station black out and subsequent loss of cooling functions for reactor and spent fuel pools (SFPs). Consequently, the fuels were damaged, hydrogen explosion blew off top of the reactor buildings and radioactive materials were released to the atmosphere and the ocean. It caused soil contamination at 1F site areas, so we had to wear coveralls and respirators when leaving the seismic isolated building to engage in work.

We have decontaminated the 1F site areas through asphalt paving, surface soil stripping, soil flipping, etc.
In order to keep 1F site areas that have been decontaminated clean and reduce the protective gear burden on workers (coveralls → work clothes), the spreading of contamination from highly contaminated areas, such as inside the Unit 1~4 buildings and the tank dismantling area must be prevented. For example, the contaminated areas must be demarcated, and the contamination of the workers, the tools, and the cars moved from the contaminated areas must be controlled.
We are now changing the policy of radiological control of areas, which enables demarcation of contaminated areas and contamination control. I'll show the approach for demarcation and contamination control.

Brief Biography:

April 2006 joined Tokyo Electric Power Company, as radiation protection section in Fukushima Daiichi NPPs
July 2010 worked in Head office as radiation protection section
October 2012 worked in Fukushima Daiichi NPPs as radiation protection section

Session 2. Primary Water Chemistry Optimization to Reduce Source Term in Belgian Units

Kim SCHILDERMANS, Patrick HEINE and Raphaël LECOCQ

ENGIE Laborelec, Belgium

Primary Water Chemistry has an important impact on the source term and correlated radiation field in and around the primary and auxiliary systems. Laborelec, as a competence center of ENGIE, provides assistance for the Belgian Nuclear Power Plants to optimize the applied primary water chemistry focussed on the primary coolant system integrity, fuel performance and source-term reduction.

The optimization of the primary water chemistry for Belgian Nuclear Power Plants regarding source term reduction is focussed on (1) primary chemistry control parameters optimization during full power operation, (2) shutdown and startup optimization, (3) identification and quantification of the surface contamination, (4) zinc injection and (5) silver and antimony issues.

1. The main primary chemistry parameters were optimized to decrease the general corrosion rate of the wetted surfaces and to maintain low corrosion product solubilities to prevent deposition and activation in the core.
2. The different stages of startup and shutdown are subjected to analyses for long term trending. Recently, optimizations were performed regarding the injection hydrogen peroxide and the usages of shutdown resins.
3. In-situ gamma spectrometry measurements are and will be performed to correlate the chemical treatment of the primary water and its impact on the activity present in the oxide layers of the primary and auxiliary system surfaces.
4. Zinc injection program has been started in Doel 3 in order to lower radio-cobalt inventory in out-of-core oxides layers and thereby also the radiation field around the primary systems. A zinc injection program will be evaluated for all the Belgian units considering different scenarios such as a possible life time extension.
5. The behaviour of silver and antimony under primary water conditions has been identified in order (1) to identify the source of both contaminants, (2) to predict its behaviour during power operation and startup/shutdown and (3) to determine how to remove the contaminant from the system.

Session 2. Elaboration of an Optimized Source Term Reduction Program for a 58 Reactors Fleet. Proposed Solution depends on the Diagnosis.

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The goal of the Nuclear Generation Division of EDF is to bring both individual and collective dosimetry results to the level of the best international standards. To reach this objective, an optimized monitoring is implemented to characterise and quantify the contamination of the primary and auxiliary systems thanks to 3 source term indicators:

- Reactor Cooling System index: it represents the state of contamination of the primary loops and of the global primary system itself,
- Reactor Building index: specified in 2011, it indicates the state of contamination of the auxiliary system and enables early diagnosis of over contamination,
- CZT gamma spectrometry measurements: it aims at characterizing dose rates. When the contamination level is too high, specific decontamination adapted to the nature and life time of the radionuclides identified (Co60, Co58, Ag110m, …) can be conducted.

The implementation of this monitoring in all nuclear power plants is one of the priorities of the ALARA action plan in order to build a relevant source term reduction program. The indicators above are analyzed to define 5 years upfront the best solutions to carry out reactor by reactor: hot spot treatment, chemical decontamination and/or optimization of the use of biological shielding.

Session 2. Ringhals Experience in Antimony and Silver related to Doses

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Background & Problem:

All together (SGR, passivation of new SGT, Stellite removal program, elevated pH and zinc injection) has lead to a situation where Co-60 and Co-58 is not any longer the dominating source for the doses. There are still a number of NPPs not performing annual or frequent surface nuclide specific measurements, leading to a lack of knowledge of the dose rate contribution from silver and antimony activation products in addition to radiocobalt. The characteristics and chemistry behavior of activations products from Ag and Sb is not very well known, nether are the main source terms, operational measures (i.e RHR-op.) and purification tools (i.e filters and resins) to minimize the negative effects of these nuclides.

Ringhals has initiated several specific R&D projects with focus to identify and understand the silver and antimony chemistry behavior and how it could be solved or optimized. The purpose of this presentation is to share with the rest of the industry Ringhals up to date data and experience and highlight the need to get together and identify industry best practices.

Distinguished Paper at the 2015 ISOE International Symposium (Rio de Janeiro, Brazil)

Session 2. EDF Feedback on the Management of Ag-110m Contamination

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In several PWR plants, it has been observed that activated silver has a significant contribution to dose rates. During reactor shutdown, activated silver deposition causes the contamination of some parts of auxiliary circuits and might impair maintenance operations. The identification of the sources of silver release, the curative treatment of silver contaminated components and the preventive measures to avoid the pollution by activated silver are three important challenges for the management PWR silver contamination. This paper refers to the EDF feedback on these issues.

EDF has defined a convenient method to confirm or to refute the fact that the source of a silver contamination is a leaking crack or a hole in the silver-indium-cadmium control rods. It has been shown that tin-113 and tin-117 activities, which are the result of the activation of indium and which could be measured in the primary coolant, are a potential tracers of a dissemination of AIC particles. The selection of these tracers is based on activation calculations and on gamma spectroscopy measurements of the coolant. We recommend the use of tin activity in the coolant as a diagnostic tool for the identification of the source of silver contamination.

As silver is preferentially deposited on cold surfaces, the contamination of parts of Chemical and Volume Control System (CVCS) and Nuclear Sampling System (NSS) can be serious. EDF has developed a chemical decontamination process which is optimized for the removal of deposited silver. This paper deals with the efficiency of this curative process through the illustration of its application at an EDF unit.

PWR contamination by Ag-110m is particularly localized onto CVCS and NSS heat exchangers. This unusual behavior is different from the behavior of the other major contributors to radiation fields such as Co-58 and Co-60. The paper proposes a comprehension of silver behavior based on gamma spectroscopy data and chemistry data. It also suggests some preventive ways to minimize or avoid silver contamination of PWR circuits.

Session 3. Neutron Detection using a Gadolinum Covered CdZnTe Detector

Romain COULON, Vladimir KONDRASOVS, Jonathan DUMAZERT, and Karim BOUDERGUI

CEA, LIST, Laboratoire Capteurs et Architectures Electroniques, F-91191 Gif-sur-Yvette, France.

***Abstract* – Gain in efficiency and miniaturization is an issue for portable neutron detectors. ^{157}Gd, ^{155}Gd and ^{113}Cd nuclei show the highest neutron capture cross-sections available in the stable element list. They are then a subject of interest for neutron detection and could be considered as suitable competitors with regards to detectors using ^{3}He, ^{10}B, ^{6}Li or proton recoils.**

A neutron detector using a Gd converter and a CdZnTe diode is studied to address portable neutron detection. To exploit the low energy signature from the Gd, a reliable compensation technique with a guard sensor has been designed. Some innovations have been done on algorithmic part and sensor part of the system. The concept has been experimentally proved. It has notably been demonstrated that a Tb cover on the guard sensor allows a reduction of the overcompensation and then a maximization of the sensitivity of the detector.

***Index Terms* — CdZnTe; Gadolinium; Neutron; Compensation; Algorithm.**

I. INTRODUCTION

NEUTRON measurement is an active subject of research driven by the necessity to find solutions to the ^{3}He shortage and portable neutron dosimeters [1, 2]. The detectors usually take advantages of high cross-section capture reactions, inducing charged ions easily separable from recoil electrons. The evolution of the cross-section as a function of the incident neutron energy is presented in Fig. 1 for the ^{3}He(n,p), ^{10}B(n,α), ^{6}Li(n,α), ^{1}H(n,n'), ^{113}Cd(n,γ), and ^{157}Gd(n,γ) reactions. Though Gadolinium and Cadmium isotopes have the highest capture cross-sections for thermal neutrons, prompt gamma rays from the radioactive capture cannot be easily discriminated from natural background gamma rays. Indeed, the capture of a neutron by ^{157}Gd forms a ^{158}Gd* nucleus within an excited state. The return to a fundamental state of energy is promptly mediated by the emission of gamma rays.

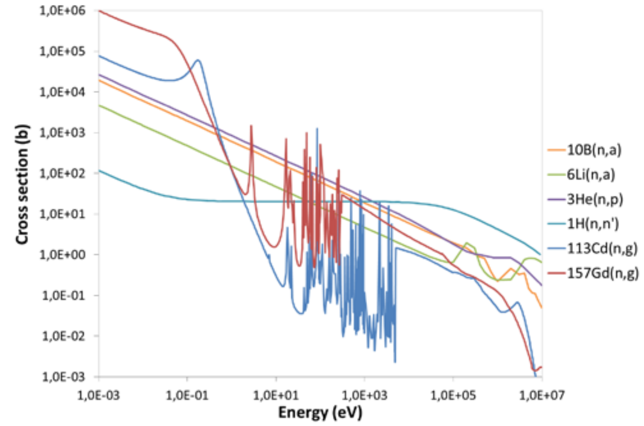

Fig. 1 Cross-sections of different isotopes used for neutron detection

However, Gadolinium-based detectors have been developed to address alternative neutron measurement. The first approach consists in loading gadolinium into scintillation detectors. In this framework, inorganic scintillators as the Gd-doped Hf02 or the LGB crystals [3-4], together with organic scintillators [5-7] must be mentioned. Robust gamma rejection and promising neutron sensitivity have been obtained in scintillation technology addressing large sensor applications [8-9].

The second approach consists in covering a detector by a gadolinium layer. Gadolinium-covered gas detectors have notably been studied in details by D.A. Abdushukurov [10]. As a representative example, S. Masaoka has developed a micro-strip gas chamber with gadolinium converters for neutron position-sensitive detectors associated to neutron scattering experiments [11].

To develop portable detectors addressing personal neutron dosimetry, semi-conductor technologies are preferable due to the high photon stopping power into condensed matter. A Gd converter has been incorporated into a MOSFET component to develop a neutron dosimeter [12]. A relationship between the output current and the neutron dose has been described. The intrinsic properties of CdTe for neutrons detection have been studied in [13]. CdTe sensors contain neutron-sensitive cadmium, which has led to the observation of a characteristic signature at 96 keV and 560 keV, resulting from neutron capture in ^{113}Cd isotopes. A Gadolinium-covered CdTe diode has been developed and tested by Miyake et al., making use of both ^{113}Cd and ^{157}Gd isotopes [14]. The small CdTe pixel detector ensures the identification of the characteristic X-ray peaks from Gd at 43 keV and 49 keV, prompt gamma-ray peaks from Gd at 79.5 keV, 89 keV, 182 keV, 199 keV,

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and from Cd at 95 keV. Signatures from Cadminium and Gadolinium have been measured but the peak area estimation is not a reliable technique to ensure a neutron metrology due to possible gamma rays interferences. A compensation system has been design and presented in the present article.

II. Source term

The source term associated with the de-excitation of a Gd or Cd nucleus following the absorption of a thermal neutron is subdivided into a prompt photon source term and a prompt electron source term. The complete equations of the nuclear reactions read:

$$^{113}_{48}\mathrm{Cd} + ^{1}_{0}\mathrm{n} \rightarrow ^{114}_{48}\mathrm{Cd}^{*} \rightarrow ^{114}_{48}\mathrm{Cd} + \gamma + XR + ICe^{-} + Ae^{-} \quad (1)$$

$$^{155}_{64}\mathrm{Gd} + ^{1}_{0}\mathrm{n} \rightarrow ^{156}_{64}\mathrm{Gd}^{*} \rightarrow ^{156}_{64}\mathrm{Gd} + \gamma + XR + ICe^{-} + Ae^{-} \quad (2)$$

$$^{157}_{64}\mathrm{Gd} + ^{1}_{0}\mathrm{n} \rightarrow ^{158}_{64}\mathrm{Gd}^{*} \rightarrow ^{158}_{64}\mathrm{Gd} + \gamma + XR + ICe^{-} + Ae^{-} \quad (3)$$

where the sum of the gamma rays labeled γ and the X rays noted XR forms the photon source term, and the sum of internal conversion electrons noted ICe^{-} and the labeled Auger electrons Ae^{-} forms the electron source term.

Prompt gamma rays produced as a result of a neutron capture into Cd or Gd nuclei are extracted from the IAEA data bank [15] and displayed in Fig.2 and 3, and internal conversion electron spectra are calculated by the BrIcc calculator [16] form Australian National University and presented in Fig. 4. Cross-sections for the most significant terms (above 1000 b) are summarized in Tab. 1. It appears that the signature of the ^{157}Gd is largely prominent. Given the size of the sensor, the signal of interest will lie below 200 keV. The measurement will specifically target 79.5 keV and 182 keV rays.

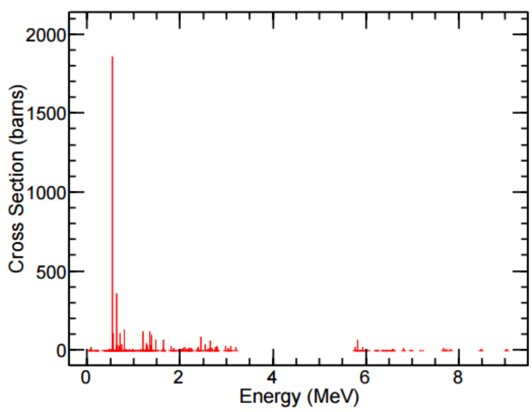

Fig. 2 Prompt gamma-ray emission spectrum of ^{113}Cd (extracted from IAEA data base [15]).

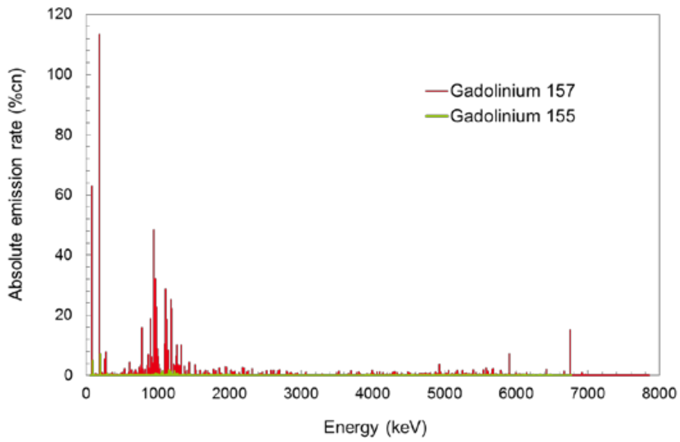

Fig. 3 Prompt gamma-ray emission spectrum of ^{155}Gd and ^{157}Gd (extracted from IAEA data base [15]).

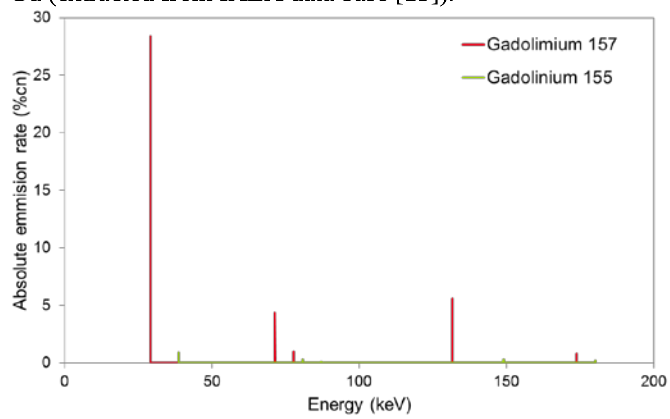

Fig. 4 Internal conversion electron spectrum of ^{155}Gd and ^{157}Gd (extracted from IAEA data base [15] and BrIcc conversion coefficient calculator [16]).

Tab. 1 Main gamma-ray and electron emissions induced by radiative capture in Gd and Cd. The emission rate is expressed in emission probability per neutron captures (%cn).

Nuclide [ref.]	Particle	Energy (keV)	Cross-section (b)	Emission rate (%cn)
^{113}Cd [15]	γ rays	558	1860	29.2
^{155}Gd [15]	γ rays	89	1380	21.7
^{155}Gd [15]	γ rays	199	2020	31.7
^{157}Gd [16]	*IC* e$^-$	29.3	1807	28.4
^{157}Gd [16]	*IC* e$^-$	71.4	280	4.4
^{157}Gd [15]	**γ rays**	**79.5**	**4010**	**63.0**
^{157}Gd [16]	*IC* e$^-$	131.7	356	5.6
^{157}Gd [15]	**γ rays**	**182**	**7210**	**113.3**
^{157}Gd [15]	γ rays	898	3290	51.7
^{157}Gd [15]	γ rays	944	3090	48.5
^{157}Gd [15]	γ rays	962	2050	32.2
^{157}Gd [15]	γ rays	977	1440	22.6
^{157}Gd [15]	γ rays	1108	1830	28.8
^{157}Gd [15]	γ rays	1186	1600	25.1

III. Method

We consider two CdZnTe diodes covered by a Gadolinium converter for the first one and a compensation cover (as Terbium) for the second one. Each diode is coupled to a charge sensitive preamplifier and a signal processing allowing counting, every time step Δt, pulses comprise between two energy boundaries. The number N_1 and N_2 of counts for respectively the reference and the guard channels have to be smoothed in order to reduce the shot noise associated with counting estimations. A nonlinear filter CST optimizing the tradeoff between response time and precision has to be implemented [17] and use to provide accurate and precise values of $\widehat{N}_1$, $\widehat{N}_2$ and their associated variances $\sigma^2(N_1)$ and $\sigma^2(N_2)$.

$$\begin{cases} \left[\widehat{N}_1; \sigma^2(N_1)\right] = CST(N_1) & (4) \\ \left[\widehat{N}_2; \sigma^2(N_2)\right] = CST(N_2) & (5) \end{cases}$$

The algorithm for the estimation of the neutron count rate $\hat{S}_n$ is based on a hypothesis test where K is a coverage factor governing the risk of false detection.

Algorithm. 1 Estimation of the neutron count rate.

$$\begin{array}{ll} If & \widehat{N}_1 - \widehat{N}_2 > K\sqrt{\sigma^2(N_1) + \sigma^2(N_2)} \\ Then & \hat{S}_n = \widehat{N}_1 - \widehat{N}_2 \\ Else & \hat{S}_n = 0 \end{array}$$

It can be noted here that smoothing is a critical point because the variance reduction will directly impact the detection limit.

IV. Experimental results

The experimental set-up is composed with a 500 mm^3 CdZnTe diode mounted on a dedicated card [18]. The diode is used successively without convertor, with a Gd convertor and with a Tb convertor. The Fig. 5 shows the diode cover by the Terbium foil. The foils of Gadolinium and Terbium have a thickness equal to 25 µm; this value is known as an optimal value as mentioned in previous study [10].

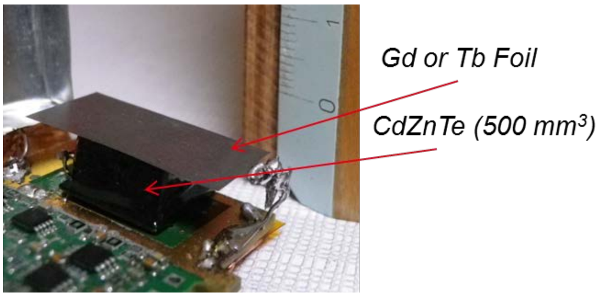

Fig. 5 CdZnTe diode covered by a Tb foil.

A source of Californium 252 is placed at 15.2 cm from the detector. A screen composed by 5 cm of Lead and 2 mm of Copper limits the gamma flux impinging the sensor. In the first configuration, a bloc of 10 cm of High Density Polyethylene PEHD is set between the source and the Lead in order to maximize the thermal neutron flux (*cf.* Fig. 6). In the second configuration, a bloc of borated wood replaces the PEHD in order to minimize the thermal neutron flux while maintaining an equivalent gamma flux on the detector (*cf.* Fig. 7).

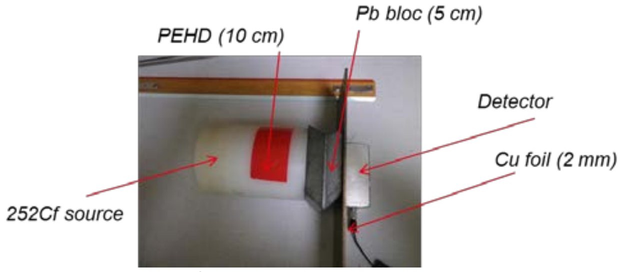

Fig. 5 Experimental configuration maximizing the thermal neutron flux.

Fig. 6 Experimental configuration minimizing the thermal neutron flux.

Spectra obtained by the subtraction of the spectrum measured respectively with the gadolinium converter and without any converters are presented in Fi.g 7. We can observe an overcompensation on the spectra below 200 keV. This phenomenon is explained by the screen effect due to the Gd foil itself. As seen in the PEHD case (red curve), the neutron increases the signal in this energy range but not enough to become significant in the range [60; 200] keV. As already observed in [19], a signal coming from X-rays at 43-44 keV is measured. To increase the signal in the range [60; 200] keV, we have then decided to cover the guard sensor by a Terbium foil with the same thickness. Terbium has been chosen because of its charge number (Z=65) nearby Gadolinium one. Fig. 8 shows subtracted spectra obtain with the use of the Terbium cover on the guard sensor. The neutron signal becomes significant with the contribution in:

- the range 70-80 keV containing the 79 keV gamma rays from ^{158}Gd* and the summation between the 29 keV internal conversion electron and 43 keV X rays,
- the range 120-140 keV containing the 132 keV internal conversion electron from ^{158}Gd* and the summation between the 29 keV internal conversion electron and the 79 keV gamma rays,
- the 558 keV gamma rays from ^{114}Cd*.

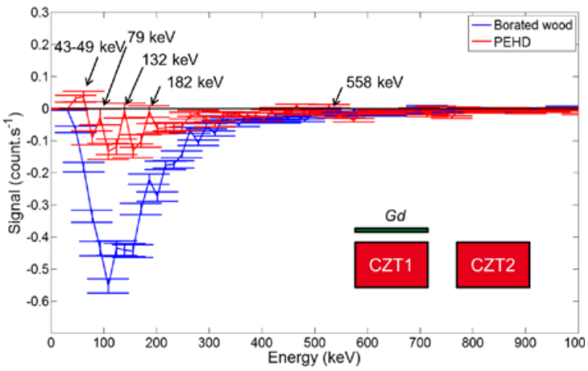

Fig. 7 Subtracted spectra measured without cover on the guard sensor.

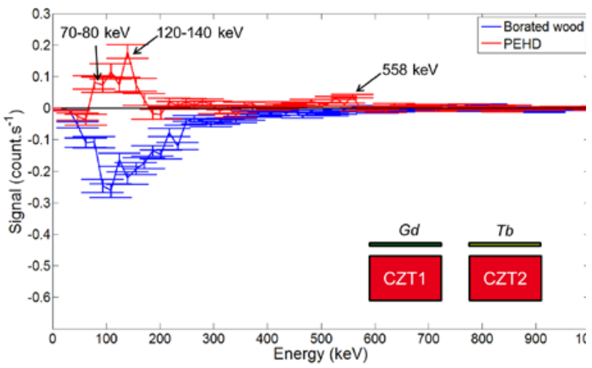

Fig. 8 Subtracted spectra measured with a Terbium cover on the guard sensor.

The Tab. 2 displays the integrated values within the range [60; 200] keV (related to gadolinium) and the range [500; 600] keV (related to cadmium). The significantly positive values measured when the neutrons are cooled by the PEHD prove the efficiency of the concept.

Tab. 2 Count rates measure in the different configurations

	PEHD Count rates (couts.s^{-1})	**Borated wood Count rates (couts.s^{-1})**
Gd / void [60; 200] keV	-0.65 ± 0.34	-3.64 ± 0.30
Gd / Tb [60; 200] keV	**+0.55 ± 0.33**	-1.76 ± 0.29
Gd / void [500; 600] keV	-0.13 ± 0.07	-0.098 ± 0.052
Gd / Tb [500; 600] keV	**+0.102 ± 0.066**	-0.070 ± 0.051
Gd / void [60; 600] keV	-1.23 ± 0.39	-4.88 ± 0.34
Gd / Tb [60; 600] keV	**+0.79 ± 0.38**	-2.45 ± 0.33

V. CONCLUSION

R&D works are in progress for the development of neutron detector based on a Gadolinium covered CdZnTe diode. This detector is particularly suitable to measure the low energy signature emitted by the radiative capture on Gd nuclei.

It has been conceived a signal processing permitting a reliable and sensitive neutron count rate metrology. Moreover the implementation of a Terbium cover on the guard sensor has permitted to reduce the overcompensation phenomenon and then to provide a significant signal coming from the prompt gamma rays between 60 to 200 keV.

The concept has been proven and future works will be engaged to optimize the design, develop a prototype and to benchmark it with competitor solutions.

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Session 3. Make Visible the Invisible for a better Radiation Protection at EDF

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Plenty of surveys on gamma cameras have pointed out following benefits from new models appearing these last years:

- Dose reduction can be achieved using gamma camera for hotspot localisation in very high dose rate areas
- Time can be saved for some systematic activities (RP mapping, transportation and waste control, …) so occupational exposure can be reduced
- Images are more understandable than notes to illustrate work documents
- Assessment of chemical decontamination and shielding

In order to improve practices in RP departments, introduction of new generation's tools like gamma cameras has been studied by EDF for several years. Tests have been performed in 2015 with industrialized products, and will be continued at the beginning of 2016. EDF compares what is available on the market to its needs for the plant: identification of the gaps and technical improvements by manufacturers when it seems possible. A call for tender will be organized in 2016, as a first step towards industrial deployment.

The objective of the presentation is to expose the different tests performed in order to assess how a gamma camera can change our practices thanks to augmented reality.

Session 3. Measurement Program for Occupational Exposure Source Terms with in-situ Gamma Spectrometry Method in NPPs

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The reduction of collective dose is a constant challenge for all nuclear power plant operators. More than eighty percent of the occupational collective doses are received during the outage for PWRs due to activated corrosion products deposited on the out- of-core surface. An occupational exposure source term characterization program was performed by CIRP and most of NPPs in China. It is very important to determine radioisotope concentrations in contaminated pipes for the reduction of collective dose. The radiation environment can be analyzed in a number of ways, among which the non-destructive measuring method would be a good choice for the operating NPPs. A radiological characterization method based on *in-situ* gamma spectrometry had been developed. The *in-situ* gamma spectrometry measuring system and the technique of virtual efficiency Monte Carlo calibration were constructed. The measuring system consists of HPGe detector, collimator, MCA and vehicle. The in situ gamma spectrometry was collected by the measuring system. The surface activity of source term and dose rate can be calculated with the Monte Carlo calibration method. As an addition to the gamma-spectrometry, contacted gamma-ray dose rates were also measured. The comparisons between calculation and measuring dose rates showed that the relative deviations were mostly within 40%. From 2005, about 18 measurement campaigns has been performed in China. The detected nuclides include Cr-51, Mn-54, Fe-59, Co-58, Co-60, Zr-95, Nb-95, Ag-110m, Sb-124.The primary contributors to out-of-core radiation fields in PWRs have been identified as Co-58 and Co-60.

Key word: in situ gamma spectrometry, radiological characterization, activated corrosion products

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Session 3. Air Sampling Programmes for Managing Internal Exposures: Review of Key Practical Issues

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ABSTRACT

Air sampling programmes are widely used within the nuclear industry to monitor workplaces. These programmes may also be used to provide quantified estimates of individual worker intakes and doses, especially where the expected exposures are low; or to provide reassurance that potential doses to groups of workers are below predefined reference levels. These programmes, if properly implemented and managed according to an effective Quality Management system, can provide a relatively simple and cost-effective means for demonstrating the radiological protection of workers; and off-set the need for more extensive and expensive individual bioassay monitoring programmes. However, there are various practical issues which need to be considered if air sampling programmes are able to achieve these aims reliably and with effective assurance. This paper provides an overview of some of the key issues which should be considered; and encompasses issues associated with sampling, monitoring, measurement, and interpretation of results. The purpose of this paper is simply to highlight these issues for information and awareness, rather than to provide detailed analyses.

INTRODUCTION

Inhalation of radioactive aerosols is the primary pathway for incorporation of radionuclides by workers, particularly in the nuclear industry. Therefore sampling and monitoring of the air in the working environment has for a long time been used as a suitable means for assessing and controlling the risk of intakes. However, despite widespread and long established use, there are still widely different approaches to the detailed techniques, methods and quality assurance by which air sampling and monitoring programmes are applied in practice. This paper provides a review of some of the practical issues which might need to be considered when establishing and operating an air sampling and/or monitoring programme: including sampling and monitoring in the workplace, laboratory methods, interpretation of data and quality management issues. The purpose is to promote awareness of these issues, rather than exhaustive detailed analyses, such that air sampling and monitoring programmes might be better optimised in terms of quality assurance and cost-beneficial operation.

SCOPE

This paper only considers sampling and monitoring of airborne particulates; it does not consider gaseous air activity. The term ‘sampling’ refers to the process of collecting airborne dust samples on a filter, which is then sent to a remote laboratory for radiometric assay. The term ‘monitoring’ refers to the collection of airborne dust samples on a filter which are subject to radiometric assay in near real time within the sample collection unit.

SAMPLING AND MONITORING: SYSTEMS AND METHODS

Personal Air Sampler (PAS)

PAS are portable, battery-powered devices worn by a worker and used to collect a sample representative of the activity concentration in the air inhaled by the worker; as such, PAS are most commonly used as a means to provide estimates of intakes by the individual worker. They are also used as a means for detecting localised acute exposures, which then trigger further investigation: e.g. involving bioassay measurements. A sampling head containing a filter is worn on the upper torso within the breathing zone, which is normally assumed to be within 30cm of nose and mouth. Ideally sampling rates should be the same as typical breathing rates for a worker (~1.2 m^3/h), but current devices often provide only about one tenth of this value. PAS devices with higher flow rates are available; however, this needs to be balanced against the ability of the device to maintain stable flowrates for continuous operation over the whole of the wear period, which is typically one working shift.

PAS might also be fitted with particle size-selective sampling heads, such that particles of different aerodynamic diameter are deposited on discrete regions of the collection filter and allow for some degree of particle size analysis [1].

Experiments with a dense monodisperse aerosol and an array of low flow-rate samplers [2] have shown that differences in sampled mass can vary by factors of around 3, even over distances less than 30 cm. This is likely to be the result of spatial variations in the bulk transport of aerosol arising from turbulent air flow. In addition, at the low particle densities typical of many nuclear aerosols, sampling statistics can lead to wide variations in sampled activity [3].

Static Air Sampler (SAS) or Workplace Sampler

SAS are commonly used to monitor general workplace conditions, to provide assurance of effective control, and to detect any deterioration in conditions. If SAS are to be used to provide information on potential worker intakes, then ideally they should be located between potential release points and likely exposure points, or occupied areas. However, in practice this can be very difficult to accomplish due to often highly complex and variable patterns of air flows within an area [4]. These air flows can be significantly influenced by many factors: building ventilation, doors and windows, ongoing operations, worker occupancy. Various means have been employed to attempt to ‘map’ airflows in a facility, with smoke tests being the more typical, although more advanced methods have been attempted [5]. However, it is advised that none of these techniques can be considered sufficiently reliable, and in all but the simplest release-exposure scenarios, there is little option but to consider multiple sampling points. The number of sampling points required will be dependent on the assessed complexity of potential release points, occupied areas and potentials for variable flows between the two. The location and number of sampling points should be subject to regular routine reviews.

Partly as a result of the problems of siting SAS, it is known from experience that SAS measurements can significantly underestimate maximum airborne activity concentrations in the workplace; in extreme cases underestimations can sometimes be of several orders of magnitude [4] [5] [6] [7]. This highlights the importance of exercising caution when using SAS measurements to infer worker intakes.

Sampling periods will need to be considered, which will be determined by various factors:

- expected magnitude of air activity
- expected risks for acute exposures or releases
- ease of access to the sampler to affect a filter change
- general environmental conditions: dusty workplaces would require high frequency filter changes to avoid accumulation of dust on the filter
- costs
- capacity of the analysis laboratory to process the required throughput

SAS devices can also provide useful information on radionuclide composition by spectroscopic analyses of the filter, and on particle size if used with a size analyser such as a cascade impactor [8].

Continuous Air Monitors (CAM)

CAMs are essentially an enhanced version of SAS which incorporate a detector facing the collection filter, connected to a real time activity monitor and alarm unit. The primary function is to provide a real time response to detect unexpected airborne releases, which would prompt evacuation of the area and remedial actions. These units might also be used like SAS for assessing chronic exposure levels within a workplace; however, it should be noted that alarm thresholds are typically set to detect acute events, and that these thresholds might not be appropriate for monitoring chronic levels of air activity to sufficiently low levels, particularly for alpha activity. In this case it would be appropriate to treat CAMs as SAS and have filter samples periodically removed for more sensitive radiometric analysis.

The accumulation of radon daughter radionuclides on the filter might present a problem by causing false alarms. Detection systems will typically incorporate some form of compensation algorithm applied to the detected alpha energy spectra – radon daughters typically having higher alpha energies than, for example, actinides. However, the detected alpha energy spectra can be significantly influenced by a variety of variable factors – e.g. dust accumulation, humidity. Therefore, even radon-compensated devices cannot be considered to be completely reliable, nor are they immune to false alarms.

Some models of CAM are available with a moving filter strip, rather than single filters. These units can be set-up to automatically move the filter media incrementally after a pre-set period of time to minimise the dust accumulation of the sampling and monitored part of the filter media. The configuration of such devices will need to be customised to the specific environmental conditions in which they are installed.

The optimum siting of CAMs might be different from the optimum siting of SAS, depending on the designed purpose of the respective sampling and monitoring programmes: e.g. a CAM might be intended to provide the earliest feasible alert of a release, whereas a SAS might be intended to provide the best estimation of potential worker intakes [4].

Sample Collection and Filter Media

Various filter media are available [4] [7] [9], the most commonly used for PAS, SAS and CAM being glass-fibre media. The primary considerations for choice of an appropriate medium are:

- Collection efficiency: this should be greater than 95% for the aerosols of interest, otherwise specific correction factors will be required to be evaluated and validated.
- Low pressure drop across the filter: if the pressure drop is too high then this could place excessive demands on the sampling pump, especially for battery-powered pumps, as are used for PAS. It could also incur excessive uncertainties for the sample collection flow-rates.
- General environmental conditions: e.g. high relative humidity could affect the performance and robustness of some filter media.

Typically, the exposed face of the sample collection medium should sample the air in the workplace, and in the vertical plane to avoid the effects of gravitational settling of larger aerosols. Where aerosol size-selective attachments (e.g. impactors) are employed then particle loss-rates within the sample head would need to be established.

SAMPLE MEASUREMENT & RECORDING

This section refers to the measurement systems and methods that are used for radiometric analyses of the sample collection filters separately from the sampling systems. These measurement systems should be located and operated within a laboratory designed for this purpose [10].

Measurement Systems

Various measurement systems are available; for the measurement of alpha and beta activity the most common systems use either proportional counter detectors or solid state (silicon) detectors. When only a low throughput of samples is required then simple single-detector manual counters might be employed; for higher levels of throughput then multiple-detector arrays and/or automated sample-changer counting systems are normally used.

System Performance & Quality Control

Measurement systems are required to be operated according to a clearly defined Quality Assurance programme to assure reliability of performance and output [10]. An effective quality assurance programme would include the following technical features [11].

Type Test: this defines the characteristics and expected performance of a system to enable the most appropriate choice of system for a particular application.

Test before First Use: this test provides assurance that the performance of a particular system conforms to the specified Type Test specification; this enables any defects or non-conformities to be identified and addressed before bringing the system into use.

Periodic Tests: this is a programme of routine periodic tests established to provide continuing assurance that the system is still performing according to the Type Test specification. These tests consider various factors as well as detector performance: e.g. mechanical reliability; software; data management; reliability of output reports.

Function check: this is a minimal check to provide assurance that a system is still functional; for air sample counters this check will normally comprise the measurement of a radioactive standard source and a background measurement.

System Calibration & Characterisation

There is no ‘natural reference matrix’ available for performing direct calibration measurements: i.e. there is no calibration source which exactly mimics the physical nature of the sample, and which contains a known and traceable radioactive content. Therefore, all calibration measurements are, by nature, indirect approximations derived from the system’s response to an analogous ‘calibration standard’. This should be borne in mind, and can cause significant uncertainties in certain circumstances: e.g. a calibration standard which is constructed of a ^{90}Sr source plated onto an aluminium substrate will give rise to enhanced beta emission rates in the forward plane due to back-scattered beta particles from the substrate; this would not be replicated by ^{90}Sr activity collected onto a glass-fibre filter.

In practice the term ‘calibration’ is often presumed to mean subtly different processes and objectives: e.g. it might refer to the initial type testing, periodic tests and/or daily function checks. The authors suggest that the term ‘calibration’ should refer to the process for determining how the system performs according to its defined Type Test specification [10]. The result will be the calculation of a correction factor(s) which is to be applied to the measured quantity to calculate the required output quantity: e.g. for the conversion of measured counts to activity. In practice for operational purposes this ‘output quantity’ will need to be expressed as various ‘derived quantities’. Therefore the ‘derived quantities’ of interest – e.g. air activity concentration – will need to be ‘characterized’ by a process which considers all appropriate issues relating to measurement of the sample, and factors pertaining to the sampling programme – e.g. dilution factors. A system can be simply checked to see if it is still performing according to its calibration by periodic quality control checks, typically by measurement of a traceable reference standard [10] [11]. Systematic reviews of the system’s characterization are usually not so straightforward and, typically, are not performed on a routine basis. However, it would be advisable for the QA programme to consider the processes for how significant changes to general operating conditions might be detected: e.g. significant changes to patterns and levels of air activity concentration, exposures to different radionuclides. This could then initiate a review of the overall characterisation of the system.

Sample Measurement

A variety of factors can affect the reliability of the sample measurement:
Particulate radon-daughters collected on the sample: this can be mitigated by using radiometric compensation methods or by delaying measurement for at least five days to allow for radioactive decay.

Radon gas within the laboratory: specific environmental controls (e.g. effective ventilation) might be required.

Differential energy response by detector: this might be an issue where the sampled radionuclides have significantly different radiation emission energies to the radiation standard source as used for calibration; correction factors might need to be considered, or the calibration and characterisation processes might need to be reviewed.

Radiation emission characteristics: factors that will need to be considered are the number and/or probability of emission of particles, and also the presence of short-lived daughter nuclides: e.g. ^{90}Sr / ^{90}Y.

Differential media substrates: radioactive standard sources are typically produced onto a metal substrate, as opposed to the glass-fibre media typically used for sample collection; this can be a significant factor for calibrating beta response due to the back-scatter of beta particles from the metal substrate, which isn't replicated for beta sample measurements.

Calibration source construction: if the radioactive element of a radiation standard source is too deep within the source then there is a risk of self-collimation of emitted particles; this might provide a forward bias for particle emissions which won't be replicated for sample measurements, where emissions are more likely to be semi-(2-Pi)-isotropic.

Differential edge effects: radioactive standard sources are typically constructed to have a homogenous distribution of the activity over most of the surface area; sampled particulates will likely have discrete deposition patterns. This issue can be overcome by employing detectors which have a surface area greater than that of both sources and samples; otherwise this factor might need to be evaluated as part of uncertainty estimations.

Self-absorption: it is feasible that sampled aerosol particulates might penetrate into the filter media, or be obscured by later accumulations of particulates. This is not generally considered a significant issue in practice for environments with low dust loading in the workplace air; however, this will need to be monitored: e.g. by periodic inter-comparisons involving full-destructive assay of samples by radio-chemistry techniques. A study was conducted at the Harwell and Dounreay sites as part of the Nuvia Dosimetry Services QA programme. This included 15 air samples (14 SAS and 1 PAS) which were exposed as part of routine sampling programmes in different facilities, at different times. In all cases the primary radionuclides were expected to be plutonium and americium isotopes. The samples were subject to the standard radiometric measurement and then sent for radiochemical analysis. The mean ratio for radiochemical to radiometric results was 1.22 +/- 0.69 (1 std dev.) [12]; which provided an indication that self-absorption is not a significant problem.

Background corrections: all detector systems will be subject to 'background'. In practice it can be difficult to ascertain the source of this 'background' and it is normally just considered to be un-defined 'noise'.

Sampling handling: sample measurements for alpha-emitting radionuclides are especially sensitive to careful handling, due to the low detection levels which are typically required. In addition to normal sampling handling requirements the laboratory should also be aware that samples collected on glass-fibre filters might be sensitive to risks of exposure to static electricity (e.g. from use of polythene bags).

NORM within filter media constructs: Some treatments of cards, particularly glazes, can include traces of naturally occurring radioactive material.

Statistical Analyses of Samples with Very Low Count Rates

When the sampled activity is very low, the count rates measured from PAS or SAS filters approximate to background counts. These counts follow a Poisson distribution [13]. This can cause problems when a background count is subtracted from the sample count. In such a case the mean of a Poisson distribution is being subtracted from members of another distribution with nearly identical mean. Since the mean of a Poisson distribution does not coincide with the median, the effect of this can be that a preponderance of negative net counts can result. This can lead to the mistaken impression that a bias exists in the background subtraction. It becomes important to ensure that the background subtraction is truly representative and that allowance is made for subtle variations in background count throughout the year. Consideration is also needed as to whether it is the mean or median of the background distribution which should be subtracted; this will be influenced by the primary purpose to the sampling – to measure individual exposure events, or cumulative exposure conditions: this decision might have a significant impact on the reported numerical values of the results.

Quantities and Units

In practice there are a variety of quantities used, and sometimes for different purposes: this is illustrated in Table 1.

Table 1: quantities and units typically used in air sampling programmes

Quantity	**Units**	**Applied correction factors**	**Typical Application(s)**
Count rate (measured quantity)	Cps, cpm, dpm, dps	Background subtraction	Type test, periodic tests, function test
Activity (output quantity)	Bq	Calibration factor	Activity on sample; Routine Lab reports and records
Airborne concentration (derived quantity)	$Bq.m^{-3}$	Characterisation factors(s) related to sampling process (eg flow rates and filters)	Health physics reports; workplace reports
Time integrated airborne concentrations (derived quantity)	$Bq.h.m^{-3}$ $Bq.s.m^{-3}$	Characterisation factors(s) related to sampling programme (e.g. coverage, sample periods, dilution factors)	Potential worker intakes
Intake rate normalised by dose coefficients (derived quantity)	DAC	Worker breathing rates, Exposure time, dose coefficients	Internal dose *rate* estimates
Total intake normalised by dose coefficients (derived quantity)	DAC-hours; DACh	Worker breathing rates, dose coefficients, default working year (hours)	*Cumulative* internal dose estimates

Uncertainty, Detection Levels, Decision Thresholds and Censored Data

The method for the determination of uncertainties in measurements is well described [14]. A thorough determination would require the construction of a mathematical model that represents all of the physical process which contribute a source of uncertainty to the final measured outcome. This would need to account for all of the effects noted in the section on **Sample Measurement**; in practice this is never attempted. It is more common to limit the quantified estimation of uncertainty to a few parameters – typically the observed count-rates from the sample, background and calibration reference standard. It can be seen form the discussions presented in the preceding sections that this approach, albeit a necessary pragmatic shortcut, can only provide a partial and qualitative understanding of the overall uncertainty in the process.

The standardised definitions and uses of the terms ‘Detection Limit’ and ‘Decision Threshold’ are generally accepted [15]. For simplicity the ‘Detection Limit’ can be summarised as the expectation of the lowest value that can be measured with a defined degree of confidence: i.e. this determination is made *before* the measurement is made. This parameter is particularly useful in planning measurement equipment, methods and procedures: e.g. the identification of appropriate equipment, sample measurement periods, tolerance levels for background rates etc. The ‘Decision Threshold’ is the lowest value of a measurement that can be claimed to be a genuine (non-background) result with a defined level of confidence. As this value is intrinsically dependent on the actual value of a measurement it can only be determined after the measured value is known. It is typical that Detection Levels and Decision Thresholds are applied to the output quantity (sample activity) or to the measured quantity (e.g. count rates). Caution should be used when using these terms applied to any derived quantity, as their determination will then need to include uncertainty estimates from all of the corrections and assumptions used in the derivation process.

Typically, standard reports and outputs will use 'censored' data: e.g. this is data that, if the measured value is less than the Decision Threshold, then the value is reported as a 'less than' value. However, there are certain applications which will require the actual measured value to be reported, with the associated measurement uncertainties: i.e. 'uncensored' data. Such reports might be required for further data analysis, particularly on bulk data: e.g. to determine mean activity levels over a period of time or a large number of samples, in which case the use of censored data would lead to biased analyses.

INTERPRETATION OF DATA AND ESTIMATING INTAKES

It is sometimes considered appropriate to use SAS measurements to estimate intakes for workers when expected doses are either low, or for confirmation that workplace conditions do not require individual monitoring programmes [16]. This is appropriate when air activity characteristics are well described and are reasonably stable; and is most often applicable for large scale processes involving low-specific activity nuclides: e.g. U, NORM. The SAS data is usually either expressed as average air activity concentrations ($Bq.m^{-3}$) which need to be modified by presumed default breathing rates and dose coefficients to derive intakes; or as time integrated exposures (DAC-hours) which need to be modified by a presumed default working year (usually 2000 hours) and by the respective Annual Limit on Intake (ALI) for the nuclides of concern. In both cases worker occupancy data needs to be recorded so that estimated intakes are proportionate to the actual time spent in the workplace. The estimated intakes will need to account for potential underestimations of the SAS measurement by the application of correction factors (sometimes known as Dilution Factors, Breathing Zone Factors or PAS Factors).

PAS are more often used for assessing worker intakes. The results of the PAS are assumed to be directly related to the worker's intake, modified by the ratio of sample flow-rate and presumed default breathing rate. A review of assessed intakes of plutonium and higher actinides at Harwell was carried out [17]. This enabled a comparison to be made between intakes assessed from bioassay sampling with initial estimates based on PAS activities. The correlation between the two estimates of intake was found to be close to zero. A similar result was found for acute and chronic intakes for individuals at Sellafield [18]. However, the same authors state that there is better agreement when groups of workers are considered. A survey of activities in routine faecal samples for groups of workers at Harwell [19] showed a clear correlation with average PAS activities for the group. There are various other studies which have attempted to compare intake estimates by PAS and bioassay measurements; however, the authors are not aware of any published studies which have attempted a rigorous (if any) uncertainty analysis for both techniques, although both techniques will be subject to substantial, but different, sources of uncertainty. Therefore any attempt to draw meaningful conclusions would seem to be of dubious value.

It is noted that there is a marked difference between sampling for plutonium and sampling for uranium. 1 DACh of pure Pu239, with an AMAD of 5 microns, corresponds to about 250 particles. Sampling statistics therefore place severe constraints on the accuracy of air sampling. However, 1 DACh of natural uranium aerosol with the same size distribution corresponds to 2.8E6 particles and sampling statistics are likely to favour more accurate estimates of intake. Air sampling data at a plant where uranium mining concentrate is converted to UF_6has been found to correlate with uptakes calculated from urine analyses [20].

Analysis of bulked PAS data for a group of workers might be useful to provide further indications of the potential probability distributions of worker intakes.

Despite the outstanding questions regarding the correlation between individual air sampling results and bioassay, air sampling still has an important role to play as a trigger for undertaking dose assessments. A preliminary survey of Harwell data [21] shows that about 25% of 81 investigations into potential acute exposures, where an acute intake was confirmed by positive follow-up bioassay measurements, were triggered by a high air sampling result (either PAS or SAS); this compares to about 50% detected by routine bioassay programmes (either urine or faeces). However, the air sampling measurements have a lower 'false trigger' rate than the bioassay programmes: i.e. an air sample trigger is more often confirmed as an intake than an investigation triggered by routine bioassay (53% compared to 29%, respectively). These analyses will be subject to further study.

OPTIMISATION: CO-ORDINATED STRATEGIES, QA & COSTS

Air sampling is not generally recommended as a means for assessing internal doses for operations where internal doses greater than 6 mSv per year are likely [16]; in such circumstances routine bioassay programmes should be considered. Where less significant doses are likely - typically in the range 1 mSv to 6 mSv per year - then air sampling might be used as the primary means for the routine assessment of intakes. However, the impact of the uncertainties implicit in air sampling programmes should be noted: these uncertainties can be significant but are often unknown and unknowable. Therefore it is general practice that air sampling, when used for this purpose, is supported by a 'reassurance' or 'confirmatory' bioassay programme [22] [23]. Such programmes need not be as extensive or frequent as those used to directly assess routine doses. The basic strategy within Nuvia Approved Dosimetry Services is to recommend 'reassurance' bioassay programmes designed to detect doses greater than 6 mSv per year. The inference from this strategy is that we claim that the 'reassurance' bioassay programmes provide confidence that doses are less than 6 mSv per year, and that the air sample programme (usually PAS) provides the best point estimate of the dose.
For operations where significant intakes are not expected – i.e. when prior risk assessments have deemed that routine intake and dose estimates are not required - then carefully designed air sampling programmes are an effective means for providing measurements to check and validate this risk assessment. However, the complexities and uncertainties implicit in the relationship between air sample measurements and worker intakes means that intakes cannot be directly quantified with any reliability. There are various approaches which might be considered to address this issue.

(a) *extended characterisation of air sampling uncertainties*: as discussed previously this is not a trivial undertaking; however, it might be cost-beneficial for large-scale processes, where operations are reasonably routine and uniform, and potential release-exposure scenarios are fairly simple and of limited scope. This might be an especially cost-effective solution where large numbers of workers are involved.
(b) *supporting 'reassurance' or 'confirmatory' bioassay*: this avoids the need for extensive characterisations; and might be more cost-effective for smaller work groups, and where exposure-release scenarios can be more varied.
(c) *task specific or campaign monitoring*: a limited bioassay study might be conducted to assess the potential intakes arising from a specific process [16] [22]. For example, at the start of a new process a limited study, involving bioassay and air sampling, can establish a partial characterisation of the air sample programme – e.g. to derive Investigation Levels. The air sampling programme then continues to monitor for any significant change in conditions; if a change is detected then the study is repeated and reviewed.

Other issues which will affect how costs are optimised also include:

- existing and available facilities, or the need to set-up from new: e.g. a bioassay laboratory will be substantially more expensive to establish from new; an extended air sampling programme will have significant costs to purchase the sampling and monitoring devices;
- technical capability of techniques & facilities for the expected exposure hazards; and also the availability and skills of staff
- knowledge (or ability to gain knowledge) of air activity characteristics: this is likely to change with time, so periodic reviews would be undertaken to see if the overall strategy needs to 'evolve';
- assessed risks of chronic and acute exposures: the assessed risks will also change with time and experience, so should also be subject to periodic reviews
- nuclide mixes (air sampling is a gross measure, whereas bioassay is normally nuclide-specific)

CONCLUSIONS

It's complicated! However, it is believed that an objective awareness of the issues associated with air sampling can lead to substantial opportunities for optimising both the effective radiological protection of the workforce, and the costs of running monitoring programmes.

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Forthcoming reference material (in preparation)

ISO 16639: Surveillance of the activity concentrations of airborne radioactive substances in the workplace of nuclear facilities

European Commission: Technical Recommendations for Monitoring Individuals for Occupational Intakes of Radionuclides

Session 3. How to Correctly Choose the List of Relevant Radionuclides to Assess Dose Uptake by Workers?

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1. Introduction

Both for day to day activities as for specific projects in a Nuclear Power Plant, ALARA studies and shielding calculations have to be performed. Such studies are based on several input parameters such as the work position, the source and the existing shielding geometries and dose rates.

For existing situations, the associated dose rate can be determined based on measurements in the field. However for situations related to accidental conditions or for new build projects, no measurement data are available. Hence, a radionuclide source term is determined based on validated models. Then, dose rates can be determined from this source term.

A radionuclide source term is a list of all possible radionuclides that can occur for the case of interest and links these radionuclides with a certain activity level. This list can include more than 1000 radionuclides of which only a small fraction have a significant contribution to the dose rate. Hence, the time to calculate the dose rate can be significantly reduced by selecting those radionuclides that contribute to the dose rate whilst neglecting those that represent an insignificant contribution.

This paper gives an overview of the methodology, developed by Tractebel ENGIE, for the reduction of the initial source term to a manageable source term. In the methodology, it was taken into account that a (slightly) conservative approach for the source term was guaranteed.

This methodology was used for several projects in the past. A short summary of some applications for the reduction of the source term is given.

2. Methodology

The methodology is based on the use of three screening steps. The first two screening steps identify the radionuclides to neglect based on radiological information such as activity and gamma ray energies. The third screening step adds some extra radionuclides to avoid neglecting at the first and second steps, some radionuclides with relatively high energy lines when shielding layers are added to the geometry.

The described methodology uses the activities defined in the initial source term 'S'. Radiological properties of the radionuclides 's' from S can be those obtained from any relevant library for radionuclides, such as the ICRP-38 publication [1], the ICRP-107 publication [2] or the ones used by the NEA Nuclear Data Information System JANIS [3].

The data that are needed are:

- The gamma lines for each radionuclide;
- The branching factor of each gamma line.

In the following paragraphs, the different screening steps are discussed.

2.1 Screening 1 – Relative activity of the radionuclides

In the first screening, the list is reduced based on the activity of the radionuclides, A_i. It is considered that only those radionuclides with sufficient activity contribute significantly to the total dose rate. In this step, radionuclides are selected based on their order of magnitude 'N'.

The factor N is determined to ensure that the sum of the activities of all radionuclides with an activity in the order of magnitude of at least N represent at least 99.99% of the total activity of the initial source term S. The selected radionuclides form together the reduced source term after the first screening, 'K'. A specific radionuclide in this list is denoted radionuclide 'k'. This is detailed in the flowchart of Figure 1.

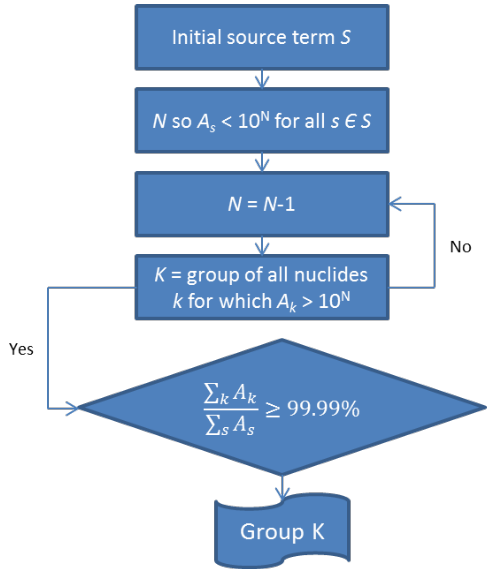

Figure 1 - Flowchart representing the first screening. Source term K is determined based on the activity levels of the radionuclides in source term S.

It is worth mentioning that the activities A_i to be considered here can take different forms, such as absolute activities (in Bq), volume activities (in Bq/m³) or mass activities (in Bq/g), since only their relative values are compared.

2.2 Screening 2 – Relative weight of the radionuclide

After the second screening, the source term K is, based on the relative weight of the radionuclides k, reduced to source term 'L'.

The weight of the radionuclide k (P_k) in terms of dose rate is determined based on the activity of the radionuclide A_k, the photon energies of the radionuclide $E_{k,j}$ and the branching factors of these photon energies $I_{k,j}$:

$$P_k = \sum_j A_k \cdot E_{k,j} \cdot I_{k,j}$$

The energies $E_{k,j}$ represent a group of photon energies of radionuclide k. Only the most penalizing gamma energies are selected. The selection of this group of energies 'J' is performed based on:

- The most abundant energies with a total representation, based on the branching factors, of at least 90% of the photons;
- The definition of the energy 'm', representing the maximum energy of the previously determined group;
- Addition of all energies higher than energy m and with an abundance (branching factor) above 5%.

Afterwards, the energy abundance of the energies in group J of the radionuclides k is rescaled to 100%.

The value P_k should be interpreted as an indication of the contribution of a certain radionuclide to the dose rate if no shielding is present.

Note that the product $A_k \cdot E_{k,j} \cdot I_{k,j}$ (in percentage) is directly given by the dose rate calculation software MicroShield [4] when selecting one single radionuclide as source term and is called the percentage of energy activity (specific to the chosen library), which eases the work.

The relative weight F_k of each radionuclide k is then determined based on the ratio of the weight P_k of the radionuclide and the sum of the weights of all radionuclides in source term K.

Subsequently, the screening consists of selecting only those radionuclides k that have a relative weight equal to or above 0.1%. These radionuclides are added to the shortlist *'L'* and a radionuclide of this group is denoted as radionuclide l. This is detailed in the flowchart of Figure 2.

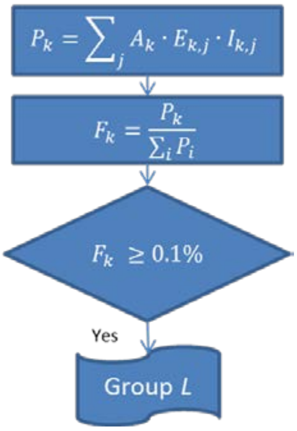

Figure 2 - Flowchart representing the second screening. Source term *L* is determined based on the relative weight of the radionuclides in source term *K*.

2.3 Screening 3 – Importance of gamma ray energy

In the third screening, the source term L is extended to take into account those radionuclides k with higher energy levels but which were excluded in screening 2. This is performed to take into account that, if more shielding layers are added, the relative contribution of high energy rays to the dose rate becomes more important. This is explained by the fact that radionuclides having a high activity and low gamma energies will be more attenuated by the shielding layers than radionuclides with lower activity but higher gamma energies. Consequently, it could occur that a radionuclide which was not selected during the second screening can still have a non-negligible influence on dose rate after addition of shielding layers in the geometry.

To determine the energy threshold, a photon energy 'M' is defined based on:

- Q_{max}, the maximum value of all the energies for all radionuclides l within list L;
- The list 'H' that consists of all photon energies 'h' (selected the same way as to determine the energy group J in the second screening) of all radionuclides l in source term L such as that $\frac{Q_{max}}{100} \leq Q_{l,h} \leq Q_{max}$;
- The energy M is then defined as the second highest energy of this group H.

As a last step, all radionuclides within group K (that were not selected in the group L following the second screening) with at least one energy that's equal to or higher than M is added to the source term L. The choice of the second highest energy of group H is a conservative approach as this increases the probability of a radionuclide in source term K to be in agreement with the criterion. This is detailed in the flowchart of Figure 3.

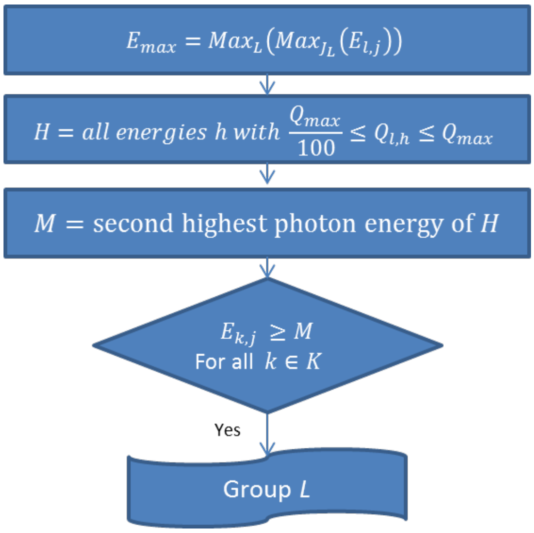

Figure 3 - Flowchart representing the second screening. Radionuclides from source term *K* are added to the source term *L* based on their radiation energies.

3. Applications

3.1 CFVS

Based on severe accident studies after the accident at the Fukushima nuclear power plant, it was decided to install a Containment Filtered Venting System (CFVS) for all Belgian nuclear power plant entities. The purpose of this CFVS system is to reduce the pressure in the reactor building in post-accident conditions by venting the containment. This air passes through several filters before release to reduce the radiological impact to the surroundings.

Due to the limited space inside the nuclear facilities, it was decided to build a new building for each entity, which will house the CFVS, in close proximity of the reactor building. However, for some of the units, the CFVS building will not be in contact with the reactor building and a gallery will be constructed to house the venting line between the reactor building and the CFVS installation.

Due to the specific use of this equipment during post-accident conditions, manual intervention will be required when operating the CFVS installation. Consequently, a study was performed to assess the required concrete wall thickness of the gallery in order to limit as low as reasonably achievable the dose rates resulting from the venting line.

The initial source term

For each power plant (Doel and Tihange), a conservative accident source term was determined using the severe accident modelling softwares MELCOR [5] and IODE (ASTEC) [6]. Both initial source terms featured 1038 radionuclides and covered 8 venting periods. For each radionuclide, an activity level, expressed in Bq/s, was determined.

As a different source term was determined for Doel and Tihange, the above discussed methodology had to be independently applied for each site. However, to incorporate additional conservatism, a single combined source term for Doel and Tihange was used for calculating the concrete wall thickness.

First screening

The first screening is performed on an individual level for each venting event. Hence, in total, the first screening was performed 8 times per entity. It was ensured that, for each venting period, the criterion of 99.99% of the total activity was complied with.

As different venting events produce different radionuclide lists, a single list, representing all the venting events, was defined. Based on the first screening, a source term *K* with 60 radionuclides and 28 radionuclides was determined for Doel and Tihange, respectively. This corresponds respectively to 6% and 3% of the radionuclides in the initial source term.

Second screening

For the second screening, only the most conservative venting event was taken into account. The most penalizing source term passing through the venting line is the first venting line as the highest activity levels are obtained during this venting event.

The selection was performed for both the Doel and Tihange short list *K* on an individual level. Based on this step, the source terms of Doel and Tihange were reduced to 23 and 15 radionuclides, respectively. These represent the source terms *L* in the methodology.

Third screening

For both source terms, the energy *M* was determined. A separate value per source term was determined but, except for those radionuclides already selected in source term *L*, no radionuclides of source term *K* corresponded to the criterion.

Consequently the final short lists of Doel and Tihange consist of 23 and 15 radionuclides, respectively. This corresponds respectively to 2.2% and 1.5% of the radionuclides in the initial source term.

As both source terms were combined, a short list of, in total, 31 radionuclides was defined. Hence, only 3% of the radionuclides were identified to have a significant impact on the dose rates outside the gallery.

3.2 EOF

One of the actions of the Belgian Stress Tests following the Fukushima accident is to build a new Emergency Operation Facility (EOF) for the Tihange nuclear power plant. This facility will be built on the site of the power plant. As the function of this facility is to support the management of emergency response after a nuclear accident in one or more of the nuclear entities on site, radioactive sources are to be expected in the vicinity of and inside the building.

The radioactive source that could be responsible for the exposure of workers inside the facility is identified as the atmospheric contamination around the building after the actuation of the filtered vent of (at least) one nuclear unit impacted by a severe accident. This source contributes to the dose rate inside the building in two-fold:

- Irradiation through the external walls;
- HVAC filters that treat external atmospheric contamination before supplying the air to the habitable zones of the building.

At the design phase of the building, keeping the occupational dose as low as reasonably achievable was set as a goal and an effective dose rate criterion was imposed in order to size the external and internal walls of the facility. This criterion was set to keep the effective dose rate below 1 mSv/h at 50 cm from the walls, which is seen as the average distance between the walls and a worker moving inside the building, and at 1.30 m above ground level, which is representative of the personal dosimeter height.

As for the CFVS application, the list of radionuclides to be considered for the dose rate calculations is large and a selection of the most relevant radionuclides is needed to make calculations manageable. The methodology presented in this work has been applied for the reduction of the initial source term. The example of the contribution to the dose rate through the external walls is detailed in the following sections.

The initial source term

In order to assess the source term around the EOF, the output of the filtered vent has been calculated the same way as for the CFVS (Tihange case), the difference being that only the fraction which is not absorbed in the CFVS installation is taken into account (mainly noble gases). Due to the distance between the CFVS installation and the EOF, a dispersion of the released activity occurs. It was considered that a uniform distribution of the activity in the volume between the CFVS stack and EOF surroundings occurs (see Figure 4).

A 1038 radionuclides source term, determined in function of time after venting time, was then obtained under the form of an activity concentration, expressed in Bq/m³, integrated during 3 minutes. It is worth mentioning that the natural leak occurring between two venting events has also been considered and added to the venting event contribution. For the calculation of wall thicknesses, the highest activity concentration in time is used.

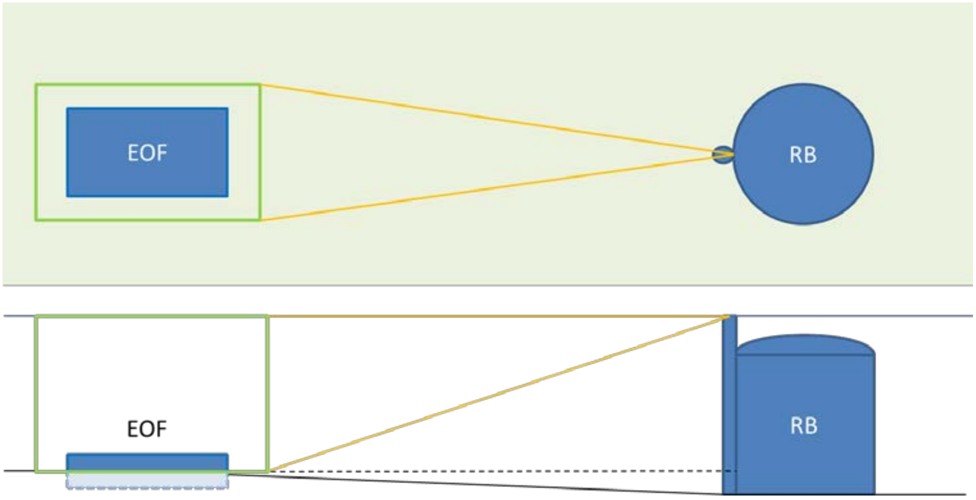

Figure 4 - The spread of activity between the CFVS stack and the EOF. In green the volume of air taken into account surrounding the EOF building. In yellow the volume of air taken into account between the CFVS stack and the EOF building.

First screening

Based on the first screening, a source term *K* with 26 radionuclides (2.5% of the radionuclides of the initial source term) was sufficient to enable reaching the criterion of 99.99% of the total activity. It can be pointed out that, in practice, 26 radionuclides is already manageable. Nevertheless, for the exercise, the next screening steps have been applied.

Second screening

The second screening, based on the weight of radionuclides from source term *K*, has enabled to shorten the list down to 8 radionuclides, which represent source term *L*.

Third screening

In order to ensure that no radionuclides featuring relatively high gamma rays have been forgotten, the third screening was applied.

Based on the determined energy *M*, three radionuclides which were not yet selected from source term *K* were added to the source term *L*.

Hence, the final short list consists of 11 radionuclides. This means that only 1.1% of the radionuclides were identified to have a significant impact on the dose rates to determine the external walls thickness.

4. Discussion

4.1 Influence of the screenings on the calculated dose rate

Based on the CFVS case, the influence of the source term reduction was determined. The following points give an indication of the sensitivity of the different screening steps. This is performed by means of MicroShield calculations based on a representative model.

Screening 1 – 'N-1'

Based on the source term of the Tihange nuclear power plant, the change in factor *N* to *N-1* (meaning considering radionuclides with an activity one order of magnitude smaller) for the screening adds 15 radionuclides to the list. However, these 15 radionuclides only represent an addition of 0.0012% in activity.

The total extra dose rate due to these 15 radionuclides was determined to be 0.018%. Hence, these radionuclides have no significant contribution in the total dose rate. Based on this result, it can be decided that the first screening does not exclude important radionuclides.

Screening 2 and 3 – With or without screening

In the second screening, radionuclides are selected based on their relative weight. A check of the difference in calculated dose rate between *K* and *L* source terms was performed.

The source term of the Tihange CFVS project was used. A dose rate simulation, identical to the simulation for screening 1, was carried out. Two calculations were performed:

- A first one including all the radionuclides of source term K (28 radionuclides);
- A second one considering only the selected radionuclides in source term L (15 radionuclides) after screening 2 and 3.

The 13 radionuclides that were excluded in screening 2 represent a dose rate fraction of 0.7%. Consequently, the conservatism of the calculation (based in particular on a conservative source term) is not influenced by this step.

4.2 Quality of the methodology

Based on the previous impact analysis of the methodology, no significant influence of the screening was detected on the calculated dose rate. The difference between the calculations with or without source term reduction is less than 1% for the considered case.

Due to the conservatism of the initial source term, the calculated dose rates remain conservative after the reduction of the source term. Even in case of less conservative source terms, the methodology developed here provides fairly good results. It is worth noting that the models (geometry of the source and the shielding) are usually also defined in a conservative way.

If a simulation is performed for the 1038 radionuclides (in the case of the two presented applications) in the initial source term, this would not be manageable. The reduction of the source term from 1038 radionuclides to 31 radionuclides (CFVS) and 11 radionuclides (EOF) limits the required simulation time a lot, without significantly influencing the calculated dose rate.

It should however be noted that if, due to new input parameters, the source term would be altered, a new screening has to be performed as the methodology strongly depends on the activities of the radionuclides in the source term.

5. Conclusion

In this paper, a methodology enabling to shorten the list of radionuclides in a source term initially composed of numerous radionuclides, down to a manageable number, is presented. This is very useful and even needed when dose rates to workers have to be assessed from a source term including a huge number of radionuclides; for instance for the assessment of dose uptake in post-accident conditions, on the basis of emergency operating procedures.

Two applications of this methodology have been discussed, where the list of radionuclides has been decreased to 3% and 1% of the initial source term.

A sensitivity analysis has been performed in order to assess the impact of this methodology on the dose rate calculation. Even if a slight underestimation can be seen (< 1%), the quality of the methodology is good, in addition of its efficiency. When the dose or dose rate calculation does not involve conservatisms (neither on the source term nor on the calculation model), it is advised to add a small conservative factor.

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Session 4. Decontamination and Post-Decon Passivation Treatment of the RRS Loops and RWCU System as a Measure for Collective Dose Reduction and Recontamination Reduction at the Cofrentes NPP in Spain

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The NPP at Cofrentes is a GE type BWR6 located near Valencia with an electrical power output of 1100 MWe. The plant went into commercial operation in March 1985 with normal hydrogen water chemistry operation NWC. Moderate hydrogen water chemistry operation HWC-M started in March 1997 after previous start of zinc injection in June 1996. In April 2010, the plant was transitioned to HWC/OLNC/Zn chemistry (OLNC™: on-line noble metal chemical addition, the process is patented by General Electric). The usual cycle length of the plant is 24 months.

The piping upstream of the heat exchangers of the reactor water cleanup system, RWCU, of the plant was decontaminated with the AREVA decontamination process CORD® CS in 2002 (RFO 13), 2005 (RFO 15) and once again in 2009 (RFO 17). The RWCU piping is made of carbon steel. After each decontamination, the piping contact dose rate experienced a fast and climbing recontamination. Transitioning of the plant to HWC/OLNC/Zn operation in 2010 did not result in the expected decrease in dose rates after the cycle (RFO 18 in 2011).

Due to the high Collective Radiation Exposure for the operational and outage personnel, accumulated during maintenance work of the past cycles, the Spanish nuclear authority CSN instructed the plant to take measures to reduce the collective dose. Within these measures, it was proposed to perform a chemical decontamination for the RFO 20, in 2015, covering the recirculation loops (stainless steel) and the RWCU system, as a huge program of maintenance and modifications on these two systems was scheduled. So in 2013, the plant operator of NPP Cofrentes and AREVA started a joint R&D program to investigate different elaborates surface treatment methods for prevention of quick recontamination of BWR reactor coolant systems after decontamination treatments. The following methods were selected: (1) Platinum deposition (Low Temperature NobleChem process LTNC™, the process is patented by General Electric); (2) Application of a self-assembling monolayer SAM (patent application of AREVA GmbH pending), and, (3) Platinum deposition followed by the application of a SAM. The methods are applied to carbon steel surfaces representative for decontaminations with the AREVA process CORD® CS. The recontamination reduction effectiveness of these methods was tested by an in-plant exposure program at NPP Cofrentes. The treated tube samples were installed in the mitigation monitoring system (MMS) of the plant, which allows the exposure of these samples to the reactor water under plant operating conditions, during the 20th Fuel cycle and removed periodically after 3 months to investigate the recontamination effects of the various passivation treatments. The passivation treatment by Platinum deposition followed by the application of a SAM showed the best results with regard to the recontamination effects. This treatment was hence chosen for the Passivation application in the carbon steel piping of the RWCU system. In addition, a post-decon Passivation treatment with optimized Platinum deposition on the stainless steel surfaces was also investigated in the joint R&D.

The decontamination and passivation treatment was performed during the outage RFO 20 in 2015 by Areva. In this presentation the decontamination and passivation application and results achieved will be described.

Session 4. Analysis of Impact of the Primary Heat Transport Purification System on Outage Radiation Fields

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The CANDU purification system uses mechanical filters to remove particulates or crud during the transients and ion exchange (IX) resins to remove dissolved impurities from the Primary Heat Transport (PHT) coolant. It is important to understand the impact of this system on the radiation fields around the various reactor components in order to minimize personnel doses during outage maintenance activities by employing the appropriate source term reduction strategy.

Outage radiation fields at the CANDU units are fully characterized by performing the Outage Activity Transport Monitoring (OATM) surveys during planned outages. This information allows estimating the changes in the radiation fields and the radionuclide composition from outage to outage. On the other hand, determining the total radionuclide activities removed from the PHT system by the purification system allows assessing the impact of the purification system on the outage radiation fields. Therefore, the gamma spectroscopy surveys of the spent filters and IX resin are performed during the replacement of these components in order to estimate the total radionuclide inventory associated with them. This data provides the most direct assessment of the amount of radionuclide activities removed from the PHT system by the purification system. Hence, the comparison between the activities deposited on the reactor components and the activities removed by the purification system allows estimating the impact of this system on the outage radiation fields.

In this paper, the novel approach for analysis of the impact of the PHT purification system on the outage radiation fields is described. The results of the in-situ gamma spectroscopy measurements of the spent IX resins and mechanical filters that were performed at Cernavoda NPP will be presented and discussed.

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Session 4. Alpha Activity and System Decontaminations

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System decontaminations have been performed in recent years at Forsmark unit 2 and 3. Both were done on the RHR and RWCU systems using AREVA'S CORD-UV process. Since both units have a history of fuel failures the amount of actinides in the resins from the decontaminations had to be determined for the waste repository inventory records. The amount of activity (including actinides) in the resins was calculated by sampling and analysis of the decontamination liquids. Pu-239/-240, Pu-238/Am-241, Cm-242 and Cm-244 were determined by nuclide specific alpha spectroscopy.

The total amount of alpha activity released from the oxide layers differed between the two units. Almost eight times more alpha activity was released from Forsmark Unit 3 than from Unit 2, even though the RHR system at Forsmark 3 had been decontaminated ten years previously, while no decontamination had been done at Unit 2. This could partly depend on differences in decontamination efficiency and partly on the history of fuel failures.

Alpha emitters, e.g. actinides, have high relative biological effectiveness and can cause large damage when ingested. On average, alpha active nuclides are 20 times more harmful than the equivalent activity of ingested beta or gamma emitting nuclides. However, the very short range of alpha particles makes them difficult to detect. When the activity of alpha emitting nuclides cannot be determined by measurements, for example in case of internal contaminations, scaling factors that describe the ratio between actinides and a key radionuclide can be used. Since the decontamination liquids also were analysed regarding gamma emitting nuclides it was possible to create scaling factors (nuclide vectors) between the activity of actinides and Co-60. Scaling factors can also be calculated from theoretical predictions of the source-term in the reactor coolant.

When comparing scaling factors based on the theoretical source-term to measured values, the amount of actinides was grossly overestimated at one unit and slightly underestimated at the second unit by the theoretical predictions. The actinides released during the system decontaminations are the result of what has been built up in the oxide layers as a result of all fuel failures in the last 10-30 years. The theoretical source-term is based on an assumption of a continuous leakage of actinides from a small fuel failure and a continuous clearance of activity.

If a hypothetical intake of actinides and Co-60 is calculated by using theoretical nuclide vectors and an assumed activity of Co-60, the actinides (here including Pu-238,-239,-240, Am-241, Cm-242, -244) contribute to 65 % of the total effective dose. Even so, the results above indicate that the dose from alphas can be underestimated if a nuclide vector based on the reactor coolant alone is used, depending on the origin of the contamination.

To conclude, analysis of alpha activity in decontamination waste shows that it is difficult to estimate the amount of actinides based solely on detection of gamma emitting nuclides (Co-60) and theoretical scaling factors. Knowledge of the plant history and specific circumstances of an incidence are needed to make appropriate assumptions to estimate the risk and calculate the effective dose.

Session 5. Radiation Protection Success of Steam Generator Replacement in Blayais Unit 3 (2014)

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In the PWRs, the Steam Generators (SGs) tubes are the place of stress corrosion cracking and mechanical wear phenomena (thermohydraulic and vibratory behavior, strain at the level of the strut slab). This phenomenon can lead to performance and safety issues for the unit itself, and may ultimately reduce the life-span of the nuclear plant.

The first strategy engaged was to plug degraded tubes. However, following several cycles of operation using this strategy, the thermal power of the SGs was reduced, outage durations and maintenance cost continued to escalate. The studies to maintain SGs in operation showed that it would be more economical to replace them than to continue a maintenance programme.

The first Steam Generator Replacements (SGRs) were carried out in the USA (Surry 2, 1979; Surry 1, 1981), Germany (Obrigheim, 1983), Sweden (Ringhals 2, 1989) and France (Dampierre 1, 1990).

The SGR in BLAYAIS unit 3, scheduled in July 2014, was the 27th of French fleets, the 21th of CPY fleets.

SGRs include an ALARA approach in charge of optimizing (reducing) the overall exposure for the whole duration of the implementation. A Working Group ALARA (EDF Study / EDF Power Plant / Subcontractor) is set up in preparation phase to define practical ways to reduce the collective and individual dose:

- Calculation of the cumulated forecast dose for the implementation, made using the estimated time for each individual tasks and the workplace local dose rate,
- Control of the radiological level of sources: precautions during the shutdown phase, purification of the primary circuit, resins change and filtrations to minimize dose rates,
- Setting up biological shielding management process,
- Water level management of the loops,
- Chemical decontamination of primary pipes ends,
- Awareness on the radiological protection of staff,
- Other good radiation protection practices...

These optimizing actions were set up to perform radiation protection in BLAYAIS unit 3 and the dose result (455 man.mSv) of the SGR in BLAYAIS unit 3 is a collective dose world record for three loops SGR.

Session 5. Reactor Vessel Head Replacement at Doel 4 – Practical Application of Dose Reduction and ALARA Techniques

Marliese VERBRUGGEN

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International experience has shown that the reactor vessel head of reactor unit Doel 4 is subject to Primary Water Stress Corrosion Cracking (PWSCC) at the level of the penetrations on the vessel head. These penetrations (control rods, thermocouples, IDD_rods) have bi-metallic welding. The material used for this welding inconel 600, and is sensitive to PWSCC.

In the preparation for the replacement, a strong collaboration with the radioprotection service was needed. The preparation consisted of composing an ALARA-file, incorporating the possibilities for dose reduction. The dose reduction can be obtained, among other things, by the strategic placement of shielding or providing additional staff to follow the project from radiological point of view.

During the preparation and the actual execution of the project, this ALARA-strategy was used for the replacement, transport of the vessel head and also the storage.

Session 5. Reactor Vessel Head Replacement at Tihange 3 – Transport Conditions and Storage of the Reactor Vessel Head

Thomas OST

Tihange NPP, Belgium

International experience has shown that the reactor vessel head of reactor units Doel 4 and Tihange 3 are subject to Primary Water Stress Corrosion Cracking (PWSCC) at the level of the penetrations on the vessel head. These penetrations (control rods, thermocouples, IDD_rods) have bi-metallic welding. The material used for this welding is inconel 600, and is sensitive to PWSCC.

The transport and storage of the reactor vessel head, complex by the nature of the equipment, its mass, the induced radiation risks and the compliance regulations has highlighted many lessons. The final storage has been realized in a building, subject to an operating license, already containing 9 steam generators and an old reactor vessel head revealed a radiation protection and compliance problem. The solutions to solve that will be detailed during the presentation.

Session 5. ALARA Program and RP Activities for the Heat Vessel Replacement in CN Vandellòs II

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During 2015, Vandellòs II NPP has replaced its vessel head. The works associated with the design modification have lasted ten months, with a total cost in Operational Collective Dose of 120,4 mSv. The main activities with significant dose impact have been the removal of the un-welded elements of the former head, the movement of the vessel head through outdoor areas up to the Spent Fuel Building where the control rod driving mechanisms (CRDM) have been cut, and a second movement for its disposal. An ALARA plan and specific RP procedures have been issued. Optimization practices have been implanted, among which the use of shielding and remote dosimetry.

Distinguished Paper at the 2015 ISOE Asian Symposium (Tokyo, Japan)

Session 5. Development of IDIS for Reducing Radiation Exposure

Dong Ug KIM

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Background

The radioactive waste which is generated from nuclear power plant is handled in order to reduce radiation pollution to workers and environment by using the pre-treatment system. The radioactive waste from plant is leaked by abnormal accident and leaked radioactive waste has a bad effect on the plant and workers. Korea Hydro & Nuclear Power Co., Ltd has tried to systemize separated drain information to use when abnormal accident like liquid waste leakage occurs and to find the causes for it.

Research and development

The liquid radioactive waste pollutes environment and equipments through drain pipes and has a worse effect on nuclear power plant and workers than normal operation. We need to review many kinds of drawings like P&ID, PLUMBING and RADIATION AREA to find the leaked and polluted area. It takes a lot of time to review all the drawings and the plant gets more polluted as time goes on. In order to systemize separated drain information, we researched pipe designs of our plant, Shin-Kori unit 3&4 and compared between the site and drawings. The pipe designs had some errors. One is that the drain pipe between radiation area and outside is linked and the other is that the drain pipe between the non-radiation area and radiation area is linked. We got drain design basis of drain pipes modified and also changed inconsistent items on the drain design basis to design company, KEPCO E&C. The drain data base was developed and integrated drawings were mapped by getting drawings above combined. Name plate of each drain hole was installed and each name plate shows the sump linked so it can be found out to which sump the liquid waste water drains.

Expectation

Shin-Kori unit 3&4 has been preparing operation. The “Integrated Drain Information System” will be the solution to reduce radiation exposure to workers by finding drain pipes faster and by setting the limits of polluted area when abnormal events like liquid waste leakage and sump level rise.

Session 6. ENETRAP III European Guidance on the Implementation of the Requirements of the Euratom BSS with respect to Education and Training for RPE and RPO

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The Euratom BSS Directive lays down specific requirements for the Radiation Protection Expert (RPE) and for the Radiation Protection Officer (RPO) which have to be transposed by each Member State into national legislation and implemented in practice. Experience has shown that, even though the specific requirements in a European Directive may be quite clear, there can be widely varying approaches to the interpretation of those requirements and implementation in practice.

It is expected that the availability of clear and substantive guidance on how the new requirements for RPE and RPO would be best implemented in Member States would be of value, not only in facilitating the implementation of the requirements across Europe, but in helping to ensure a consistent approach.

This guidance document has been developed within the framework of ENETRAP III WP7 “Guidance to support the implementation of E&T requirements for RPE and RPO as defined in the Euratom BSS”. The objective of WP7 activities is to facilitate the implementation of the new requirements for RPE and RPO in Member States and to help ensuring a consistent approach throughout the European Union.

In this guidance document all key issues for RPE and RPO are addressed:
- adoption of requirements into legislation;
- intended roles/functions/duties of RPE and RPO;
- required infrastructures and mechanism for recognition (RPE);
- suitability and competence requirements (RPE and RPO);
- appropriate education and training.

The guidance proposed will complement the guidance being developed in the medical field by facilitating the implementation of the new requirements for RPE and RPO in Member States and helping to ensure a consistent approach throughout the European Union.

Session 6. Challenges of the Aging of the Brazilian Regulatory Authority

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Ageing is not a subject to be considered only to equipments and machinery. In any organization Human Resources also show the effects of ageing. Usually, as people grow older, they also become more experienced and develop a tacit and useful knowledge that is applied in specific fields. The Brazilian National Nuclear Energy Commission (CNEN) is facing this issue already. The retirement of several senior in the past few years and the expectation of retirement of roughly half (18 seniors) the remaining seniors in the next ten years has lead the General Nuclear Reactors Coordination (CGRC), and other sectors of CNEN, to seek replacement of some seniors.

In order to achieve continuity in the regulatory works CNEN has to recruit, and train, new professionals to occupy the existing and projected vacancies. The recruitment process to Brazilian public institutions is not really straightforward. The past recruitment processes to the reactors licensing area happened in 2002, 2010 and 2014. The last group joined CGRC in 2014/2015 and consists of ten new workers and 2 transfers from other sectors of CNEN. This group have a young profile with most of them arriving from areas not related with nuclear reactors. The challenge that CNEN face now is how to train this group within the necessary tacit knowledge before the seniors retires. Initially the whole group received internal induction training on several subjects related to nuclear reactor technology, but each area involved need a specific training on their own.

The occupational and environmental radiation protection area, for instance, is following two different approaches to this problem. On-the-job training, following the senior professional in inspections on NPPs and research reactors and formal training with experienced professionals from other institutions (national and international) when available. In order to keep up with the standards wanted by the regulatory body the system must provide a way to keep with a continued training program. Either having a formal way to reach new postgraduate degrees, or having specialized training related directly with the expertise field of each professional.

Session 6. Integration of Human-Factors within the Design of a Training Tool within the Radiation Protection Field: an Opportunity to Foster the Tool's Appropriation and Develop the Trainer's Competences

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The R&D EDF project RODIN (Gathering People to Optimize Doses on Nuclear power plants) aims to design tools and methods to improve workers' radiation protection on EDF NPPs. In particular, a numerical radiation protection training simulator (CERNum) is developed to simulate the radioactivity in an environment which is representative of the industrial background (Courageot & Kutschera, 2014). This technical development aims, at the end, to enhance workers to adopt risk prevention behaviors and risk detection practices in existing work situations. On one hand, we rely on the general postulate that to design a useful training device and relevant training contents, the work and social context of the targeted populations (Mhulmann, 2001) have to be characterized. On the other hand, we believe that the use of new technologies opens up new ways to improve and develop learning simulations, (Parage et Bomal, 2015).
The CERNum design process integrated both engineering and social sciences very early in the process to carry out three main objectives in order to design: 1) a human-machine interface adapted for the trainers work, 2) contents of training meeting the challenge of risk detection and management of drifts conditions, 3) professionalization path for trainers which go for bringing them to use active learning method which is based on the responsibility of learning on learners (Bonwell & Eison 1991) and experiential learning. "Active learning" method postulates that to learn, people must do more than just listen: they must read, write, discuss, behave, or be engaged in solving problems and experiential learning or "learning through reflection on doing" postulate that people to learn should experiment, test, investigate solutions (Lewis & Carol, 1994). The global objective of the human factors integration within the design process of CerNum aimed to transform this simulation tool, essentially viewed as a technical evolution designed by engineers, in a new set of professional resources (device, script based on real work, learning methods) for trainers to develop their competences in thinking, writing "learning script", animating the simulation and the debriefing.

This communication proposes a reflection about the consideration of human-factors aspects in upstream phase of the design process of an innovative training device in the field of radiation-protection. This reflection results from a set of studies which are leaded within a collaboration between EDF R&D and an academic research laboratory CNRS- LIMSI.
In this research program, the integration of human-factors is based on a works analysis approach of three different populations: trainers, trainees and workers. The analysis presented are supported by two theoretical frameworks: ergonomics and professional didactics (Boccara & Delgoulet, 2015). The works analysis approach results from observations in the "real life" of trainers, trainees, workers and interviews. Our approach must allow to identify and to take into account baseline situations (Samurçay & Rogalski, 1998) which are relevant to identify probable future training and working situations. The challenge is to understand and to anticipate, together and in the same time, the trainees and trainers activities, very early in the design process in order to contribute to the development of the new device (Daniellou, 2004) and to identify possible evolutions of the teaching and learning activities with the introduction of this new device (Horsik, 2014; Faning & Gaba, 2007). This approach adopted by researchers feeds a participatory design process (Beguin & Cerf, 2004; Barcellini, Van Belleghem, & Daniellou, 2014) centered on the anticipation of the uses and the identification of the future probable situations to direct and act during the process of designing the new training tool: CERNum. Depending on the step of the participative method, this approach involves engineers, radiation-protection experts, radiation-protection practitioner such as the final users: the trainers and the trainees. The participation of these various

actors was consequently structured to foster the appropriation of the CERNum by trainers from the upstream phases of design process.

These multidisciplinary approach will be discussed and based on the three main objectives, mentioned above, in order to point out the efficacy conditions which facilitate the integration of an innovative training device within an industrial context.

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Session 6. Radiation Protection Planning as a relevant Course Content for Training ALARA in NPP

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For education and training of ALARA in RP, exercises on drawing up a radiation protection plan are relevant component of the RP courses in Switzerland. In the courses, teachers, who are mostly well-experienced RP professionals from NPP, are demonstrating RP plans from the real work floor including the experience gained when putting the plan into practice. The course participants have to exercise and discuss several examples of RP in detail. Especially for RPE (Strahlenschutzsachverständige) and RPO (Strahlenschutztechniker und –Fachkräfte) the necessary aspects of a RP plan, including the optimization of protection measures, are trained through working in groups on a given simple example for radiological relevant assignments. In the report, I will explain one example in detail showing how to teach aspects regarding ALARA culture. In some courses, the final examination comprises drawing up of a RP plan of a complex job in an outage or of getting control after a malfunction or an incident.

Session 7. Radiation Protection: an Approach for Simplification

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Key word for EDF in order to face the increase of maintenance activities and improve processes is "Simplify". Plenty of measures have been set to reach optimized operations and Radiation Protection management, and others are still to be developed. Following fields are part of the simplification project:

- Reference standards
- Performance management
- Documents (guidelines for significant works)
- Practices standardization (ALARA committees, hot spots management, chemical decontamination, shielding, …)
- New tools to facilitate activities in controlled area (automatic distribution of radiameters, RP mobile application, …)

The objective of the presentation is to expose the different measures already implemented to facilitate RP at EDF, and those still to be instructed. Two projects will be developed to illustrate the gain in terms of simplification and facilitation:

- Distribution of RP and safety equipment on the way of workers going to Controlled Area: Radiabox are being installed in all changing rooms of NPPs. Administration of the machines is available from the network.
- Reference standards' simplification: some themes such as work areas, radioactive sources, RP management, etc… have been reviewed to make things easier for workers and organizations.
- RFID Radioactive source management for gammagraphy: greater availability, autonomy and traceability.

Further measures to be instructed concern practices to standardize between the 19 NPPs. Moreover, EDF is looking for limiting local additive prescription on site.

Session 7. Demonstrating ALARA across the UK Nuclear Industry - Implementation of the Control of Occupational Radiation Exposure (CORE) Regulatory Inspection Project

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The Office for Nuclear Regulation (ONR) is undertaking the Control of Occupational Radiation Exposure (CORE) project in order to assure itself and its stakeholders that doses incurred by workers in the UK nuclear industry are ALARA. The project requires the inspection of 37 nuclear sites across the UK within the three-year duration of the project, with approximately half of the site visits completed to date. In addition to assessing the compliance of nuclear sites with relevant UK legislation, the project is intended to identify any industry-wide themes that could be considered to be areas for improvement and to enable the identification and sharing of examples of good practice between nuclear sites. We present the process that we have followed and a summary of the findings that have been identified to date, including potential areas for improvement and examples of good practice.

Session 7. The Way to Optimize Radiation Exposure Index at the Russian Nuclear Power Plants

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In actual fact, it was 20 years before, when the Federal Law on "Radiation safety for Population" with more rigid limits for personal dose rates was established in Russia.

Thanks to implementation of set of managerial and technical measures during the period of the years 1995-2014, a three times decrease of radiation exposure on personnel was reached. During the latest 10 years the rate of radiation exposure drop-off slowed down, and actually all nuclear power plants reached the certain level around which there occurs a fluctuation, depending on the scope of radiation hazardous operations.

Considering the current doze rates at the plants, it should be noted that basic doze limits are observed by all NPPs. More than 90% of personnel obtain individual dose less than 5 mZv.

It is also worth mentioning that at radiation exposure values at VVER NPPs are similar to the same indexes at the foreign nuclear power plants. Because of the design specific features, maximum individual and collective doses are typical to RBMK channel type plants.

With consideration of the reached level of radiation safety at the nuclear power plants, the following goals for the nearest future may be set forth:
- reducing the "critical group" of personnel;
- upgrading radiation situation and decreasing radiation exposure at the channel NPPs;
- optimizing radiation exposure at the pressure vessel reactors.

Solving these and other tasks in the area of ensuring radiation safety at the Russian nuclear power plants will in many ways facilitate implementation of measures, stipulated by the developed and implemented work programs, and in particular "Optimization of personnel radiation protection program".

Session 7. Improving Occupational Radiation Exposure using ALARA Tools: Performance Indicators

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ABSTRACT

Over the last 8 years, many administrative and procedure level measures, have been implemented, to improve total collective dose. The most important global result is reducing station dose from 520 man mSv in 2007 to 388 man mSv in 2015, for two units.

Involving working group ALARA coordinators in planning and tracking exposure is contributing in achieving both individual and overall department goals. The excellent results in collective radiation exposure have been obtained also, by improving working group good practices.

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

Radiation protection department permanently monitors workers performance inside radiological area. All radiation protection deficiencies are daily analyzed and rapid corrective action are implemented, if necessary. Periodically (quarterly), radiation protection trend analyses are performed to monitor the evolution of the radiation protection deficiencies. 12 categories of deficiencies including: radiation protection work practices (monitoring, contamination control, rad waste collection), contamination control (both personal and material), use of protective equipment, RWP/work planning and ALARA performance indicators have been identified and followed. The most significant improvement has been achieved in radioactive materials control, after implementing a corrective action plan when number of events decreased from 15 events in 2nd quarter 2013 to none in 4th quarter 2015.

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

1.0 INTRODUCTION

CNE Cernavoda management is committed to continuously improve the safety standards in order to protect personnel, public and environment. ALARA is an important element of the global approach to radiological protection and plant management commitment to ALARA has been clearly stated by the reference document "Radiation Protection Principles, Policy and Regulation".

Keeping exposures ALARA is first a way of thinking, rather than a formula. It is very important for radiation protection personnel to collaborate closely with working groups and make them aware for being responsible for doses they received. Radiation workers, ALARA coordinators, first line supervisors, and managers are directly responsible for controlling and reducing radiation doses.

During 20 years of operation, most of the exposures were below the Recording Level and the majority of recordable doses were less than 1 mSv. No legal or administrative individual dose limit has been exceeded. The actual levels of individual and collective effective doses due to external and internal exposures reveal the effectiveness of implementation of the Radiation Safety Policies and Principles established by the management of the Cernavoda NPP. Despite the increased number of professionally exposed workers after starting the operation of Unit 2 in 2007, the collective doses did not increase accordingly.

Collective dose provides indication about plant radiological condition in connection with personnel behavior. Best dose performance is a result of a balanced combination between those two factors, which represents the philosophy of ALARA principle. ALARA principle has much qualitative and less quantitative connotations, that's why its implementation is opened to several methods or particular approach.

First level of implementation is using microALARA techniques converted into radiation protection measures for every radiologic risk activity. An aggressive policy to reduce individual exposure was applied since 2005, including:

- a strict control of D2O leaks and leaks reduction program
- providing dryers availability
- optimization of personnel access in R/B
- using appropriate RP protective equipment
- hot spot management program
- implementation of RWP system

Collective dose improving results needed also an improved personnel behavior inside radiological area. And to improve human behavior, we needed to find out the exactly near-misses made during activities. Two directions were considered with the aim of correcting personnel behavior: define ALARA performance indicators for working groups and elaborate periodically trend analyses for radiation protection deficiencies.

In the beginning trend analysis addressed only human performance deficiencies with direct and significant radiological impact. The results of those analyses were corrective actions for causative working group staff.

Analyzing abnormal conditions reports on radiation protection violation, we extend this trend analysis to include any kind of deviation from RP procedures. Currently trend analysis include: quarterly value for strategic performance indicators, monthly distribution of abnormal condition reports grouped by type of deficiency and number of deficiencies evolution for the last four quarters.

2.0 ALARA PERFORMANCE INDICATORS

After implementing ALARA and RWP programs, a continuous station focus on collective radiation exposure reduction has resulted in top industry performance for CANDU designed reactors over the last 8 years, reducing station dose from 271 man mSv / unit in 2007 to 194 man mSv / unit in 2015.

The station's exposure control program continues to be in full compliance with the regulatory requirements. In particular, the station exposure control level of 14 mSv/calendar year is below the single year regulatory limit of 20 mSv / year.

The main indicators to be looked at are the collective dose and the distribution of individual doses. The targets for these indicators are obtained through a generic description of the major radiological jobs that are planned to be performed (based on a rough estimate of the frequency of the jobs performed, their duration, dose rates and number of workers exposed) correlated with statistical (historical) values.

After three consecutive years - 2004, 2005 and 2006 - of major concern on individual and collective internal doses (contributing with up to 60% to the total dose), due to the increase of tritium dose rate in the Reactor Building, important steps were done to decrease this type of exposure.

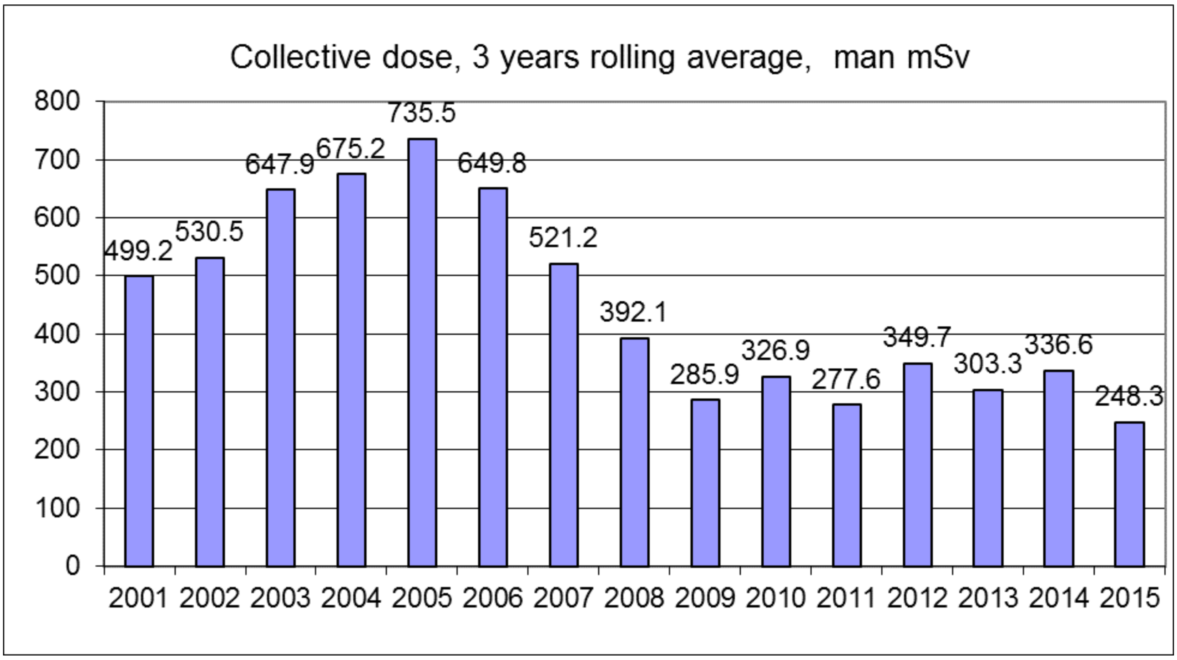

Fig. 1, Collective dose, three years rolling average

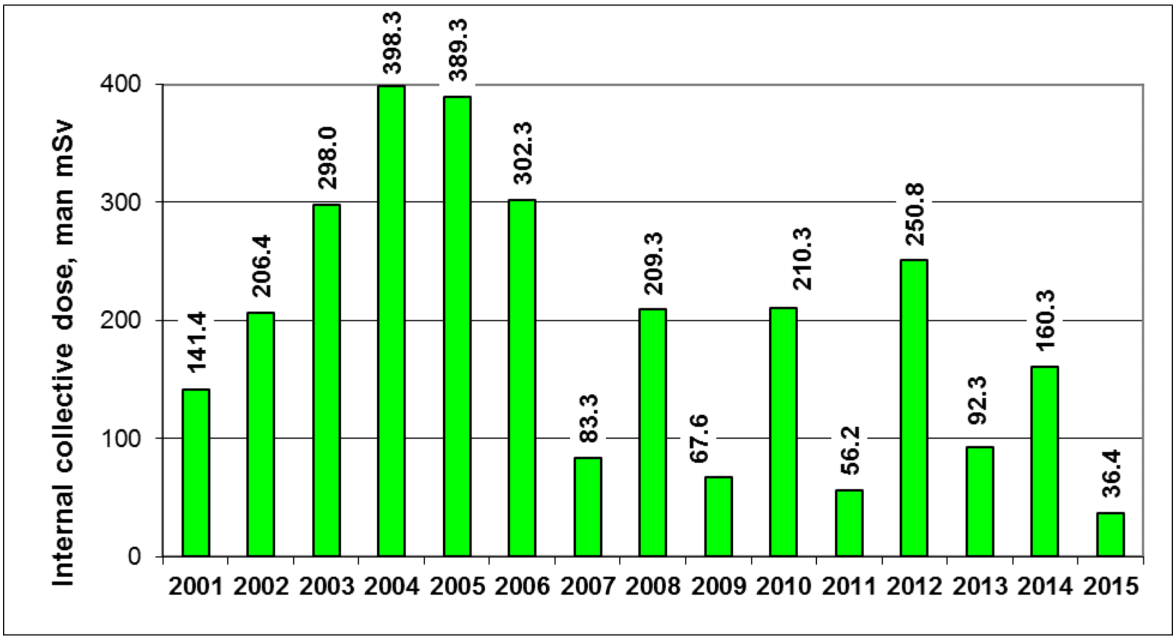

Fig. 2: Internal collective dose

An aggressive policy to reduce tritium exposure was applied since 2005 including:

- a strict control of D2O leaks,
- providing dryers availability
- optimization of personnel access in R/B,
- using appropriate RP protective equipment.

Corrective and preventive actions and recommendations, aiming both work planning (exposure control) and technical aspects, worked efficiently.

Bad performance of both collective and internal doses became triggers to decide implementing of exposure reducing policy, more interactive and in relationship with working groups. In addition with pre and post job activities evaluation, a set of performance indicators was defined, in order to closely monitor the personnel behavior related with radiation exposure inside working area.

3.0 NEW ALARA PERFORMANCE INDICATORS

Starting with 2008 we operated two units, and in its first meeting in 2008, ALARA Technical Committee approved challenging values for collective dose and internal dose contribution: 688 man mSv (with Unit # 1 planned outage) and 30 %, respectively.

In order to further improve plant performance related with exposure of radiation workers ALARA committee approved the implementation of some new performance indicators for the major work groups and for the plant:
- Unexpected acute individual external exposures;
- Unexpected acute individual internal exposures;
- Maximum individual dose;
- Internal contaminations with radio-nuclides other than tritium;
- Unexpected contamination of surfaces;
- Personnel contamination identified at the exit of the RCA.

2008 dose results confirmed the first good steps of ALARA principle implementation. Station collective dose and internal collective dose started to decrease and plant management commitment to ALARA has been clearly stated by the reference document “Radiation Protection Policy and Programs”. The awareness of Radiation Protection in the station became a topic in planning meeting agenda. Twice a month, collective dose distribution by working groups is presented to plant management, including senior supervisors of working groups, who can analyze spent dose budget versus monthly target.

Also, first ALARA Annual Report has been issued for 2008 to present station ALARA performance. This report pictured the exact state of radiation programs efficiency and areas to be improved have been identified.

In the following years we defined a few more ALARA performance indicators based on EPRI and WANO/INPO guides. In present, 10 ALARA performance indicators are monthly reported and analyzed with working group ALARA Coordinators. New 4 (four) ALARA performance indicators are:
- Inadequate response to EPD’s dose rate / dose alarms
- Maximum individual internal dose
- Personnel Contamination Events (inside Radiation Controlled Area - RCA)
- Unexpected exposures: external and effective over 0.1 mSv

They are assessed and reported periodically to reflect the objectives and permanently mark out achievements and breakdowns. Depending on the performance, every year all performance indicators values are analyzed and they could be redefined or targets readjusted to reflect the efficiency of professional exposure control process. Though, radiation protection ALARA personnel record every deficiency and investigate all events which exceed target values. Corrective and preventive actions and recommendations aim both work planning (exposure control) and technical aspects, so that work conditions (especially radiation work) to be improved.

Five years dose reduction plan has been developed and approved by senior management to provide oversight and resources for dose reduction initiative.

After 2010, ALARA reports showed a negative trend for “Unexpected acute individual external exposures”, measured by personnel response to EPD dose alarm. Event analysis revealed weaknesses in applying EPD dose alarm response, even when worker knew his PAD alarm had been activated. To improve this performance has been necessary to come closer to the workers, identify steps with radiation protection impact and give them support to avoid unnecessary exposure.

4.0 TREND ANALYSIS RESULTS

Integrated root cause analysis for inappropriate response to EPD dose / dose rate alarms opened our vision to identify all radiation protection deficiencies, group them against causal factors, follow the trend and make conclusions or corrective actions whenever are needed.

Since 2013, quarterly trend analysis are made and discussed with working groups from Production Division during radiation protection and industrial safety meeting.

Radiation protection deficiencies identified till now, are:

1. Heavy water leaks
2. Use of RWP and work planning
3. Contamination control
4. Radioactive material control
5. Radiation protection work practice
6. Inappropriate EPD alarm response
7. Adverse trend for ALARA indicators
8. Protective individual equipment deficiencies
9. Radiological conditions
10. Work practices during high radiological risk activities
11. Personal contamination events
12. RP fundamentals

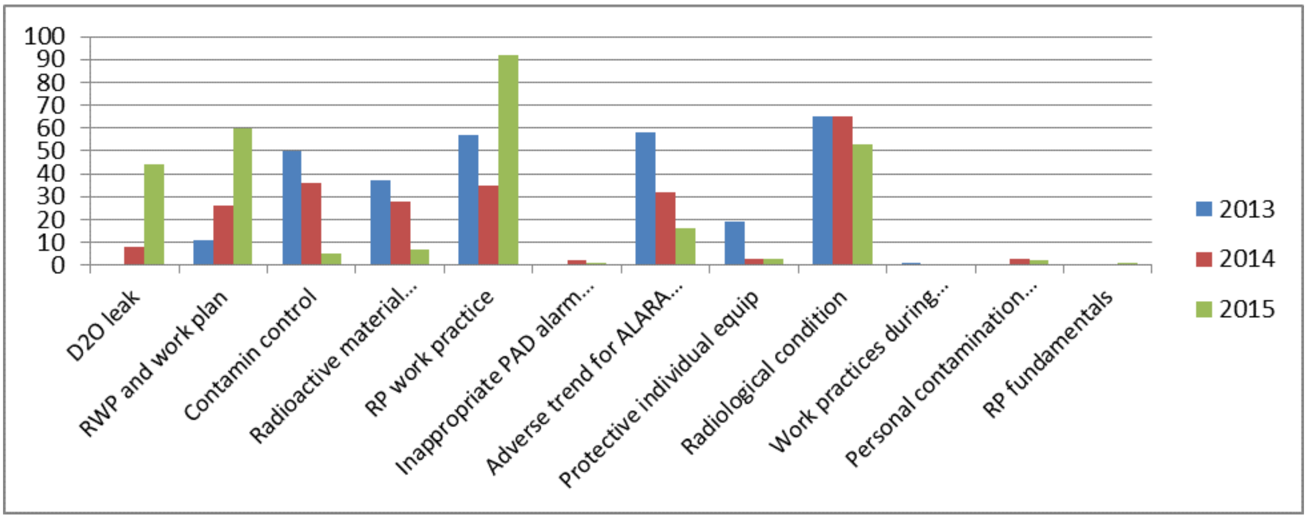

Fig. 3 Evolution of radiation protection deficiencies

A three years picture of radiation protection deficiencies shows improvements in contamination control, radioactive material control, adverse trend for ALARA indicators and radiological condition. Contamination control has been a hot issue in 2013 and RP department developed an action plan to reduce number of violations of procedure and this theme is annually analyzed into a focused self-assess report. Same idea was applied also to reduce radioactive material control deficiency in addition with many observations and coaching sessions developed for this topic.

"Radiological conditions" and "Heavy water leaks" deficiencies accounts for equipment defects generating increased radiation dose rates. In order to improve radiological conditions, radiation protection department implemented a program for hot spot management and upgraded tritium in air monitoring system in Unit #1. Also, we made efforts to accelerate implementation of support system with good impact in radiological condition: portable dryers, installing air dehumidifier in reactor building. These deficiencies are carefully followed in order to identify necessary systems improvements and the efficiency of leaks management.

Starting 2015, during monthly meetings, Technical ALARA Committee analyses from radiological point of view every job if a difference higher than 25% between estimated and received doses is registered. This is considered a radiation protection deficiency and, if necessary, corrective actions are established in order to improve the performance.

Three more categories were added as new type of abnormal condition occurred: D2O leaks, personal contamination events and RP fundamentals. To correct personnel behavior it is under implementation a RP fundamentals training course as a prerequisite of periodical radiation protection skills testing. RP Fundamentals for RP personnel and workers will be reinforced with special attention to high radiological risk jobs, in particular radiography and high activity materials manipulation. Radiation protection themes are included in "Subject of the week" training materials of Operation and Maintenance Departments.

The awareness of Radiation Protection in the station and ownership of dose have been increased by placing in key high traffic areas of the plant specific information: charts, bulletin, newsletter on RP stations goals, ALARA initiatives, RP policies and procedures.

5.0 CONCLUSIONS

ALARA is an important basic principle of radiological protection. Keeping exposures ALARA is first a way of thinking, rather than a formula. It is very important for radiation protection personnel to collaborate closely with working groups and make them aware for being responsible for doses they received. Making radiation workers, ALARA coordinators, first line supervisors, and managers directly responsible for controlling and reducing exposures open up for better dose performance.

ALARA performance indicators are useful if they are used to identify the low level errors generated by poor radiation protection working practice with exposure consequences. RP personnel grant support and coaching for high radiological risk, but worker alignment are important to achieve exposures that are kept ALARA. Since the objective of the optimization of radiological protection is to keep individual and collective doses below the appropriate dose constraints, the most relevant indicator is the dose (collective or individual). Good results for dose are the outcome of good adherence to the radiation protection procedures.

Making periodic trend analysis of RP deficiencies and related corrective action plans could also contribute to performance improvement. This is demonstrated by our improved performance for contamination control (50 deficiencies in 2013, 5 in 2015 and 0 in 1st quarter of 2016) and radioactive material control (37 deficiencies in 2013, 7 in 2015, 2 in 1st quarter 2016).

There is still room for improvement, we must stay focused on “Radiation protection work practices” (57 deficiencies in 2013, 92 in 2015 and 27 in 1st quarter 2016), particularly on: monitoring equipment used in zone 1, RP individual equipment abandoned in the work areas (half masks, gloves), temporary storage of contaminated materials, arrange and working in rubber area.

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Session 7. The Management of Radiographic Controls on the Construction Site of the Flamanville 3 EPR Reactor

Franck VEYSSIERE (RP Officer)[1], Stéphane LELONG (RP and Safety Manager)[1], Gonzague ABELA (RP Expert)[2]

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The Flamanville 3 is the largest construction site in Europe, with more than 4,000 workers on one unit.

The number of annual radiographic Non Destructive Testing was 46,000 in 2015.

European and French regulations assigns responsibility for coordinating the work, including radiographic controls, to the EDF project. The radiography company is responsible for its activity including radiation protection workers handling radioactive sources.

EDF imposes a level of radiation protection on the Flamanville 3 site at the same level as the operating units, but the defense lines are not yet all present on this site, including:

- Detection devices of radiation related to operations
- Individual dosimeters in the nuclear island
- Radiation protection culture of the workers who have not all been worked on operating EDF sites.

The EDF construction management of Flamanville 3 has set up a specific organization to control this risk.

The presentation describes the organization since the arrival of the sources on the site until their departure. The main steps of radiographic checks being:

- Planning meetings
- radiographic inspection permits issued by EDF
- The field visits before the controls
- A coordination meeting with all team leaders early in the night shift
- The presence of coordinators to check compliance with the rules, markup and evacuation of areas of operation before the controls start.

The incidents are plotted and analyzed jointly by EDF and radiologists. The Nuclear Safety Authority is kept informed of the events.

Session 8. Implementation of the Radiological Protection Principles in Decommissioning: Lessons Learnt from Three Projects

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Since the last decade of the 20th century, the Belgian Nuclear Research Centre (BNRC) of Mol in Belgium is involved in decommissioning projects. In the early nineties and partially supported by the EC, the Centre launched the decommissioning project of the first European PWR BR3, whose main objectives were to compare the cutting techniques and to implement the ALARA principle. The experience built by the Centre during this project was recognized and as a consequence, the Centre has been involved in two other decommissioning projects: a research reactor at the Gent University premises and the MOX production facility Belgonucléaire. Although the levels of completion of these three projects are slightly different, they already allow for identifying the main lessons learnt from the implementation of the radiological protection principles, with particular emphasis on the optimization principle. These lessons as well as those learnt from the global risk management in three different nuclear installations under decommissioning have been used – among some other decommissioning projects in other countries – during the development of a new IAEA TECDOC which will be published in the coming months.

Session 8. Optimisation of RP during the Course of the Decommissioning of Spent Fuel Channels

Stephan B. NAVERT

BKW Energie AG; Kernkraftwerk Mühleberg; CH 3203 Mühleberg

In BWR, the fuel channels are elements of the reactor core which guide the cooling agent through the fuel bundles to optimize the heat transfer from the fuel to the water. These fuel channels are in the core throughout the life of a fuel bundle (in average 6 cycles) and thus get highly irradiated and activated.

Decommissioning activities include the shredding, compacting, packaging and transfer of the resulting waste packages to the on-site intermediate waste storage facilities. Due to the high dose rates from the fuel channels, most of the activities is handled remotely under water or in shielded operation boxes. However, technical problems can result in interventions which could result in elevated radiation exposures to operators and RP-personnel.

The presentation will demonstrate the various improvements that have been identified and implemented over of several campaigns, together with their impact on the radiation exposure to personnel.

Session 8. José Cabrera NPP - Impact of the Chemical Decontamination on the Decommissioning Works. Current State of Decommissioning Works

José CAMPOS MENDIA

ENRESA, Spain

Introduction

The full system chemical decontamination process was developed in 2007, three years before starting the decommissioning works.

This process results in a fundamental ALARA factor to reduce collective and individual doses in the current decommissioning Works.

Main objectives of the process are:

- To reduce dose rates, and in consequence, reduce individual and collective doses
- To reduce airborne contamination levels and therefore personnel intakes risk
- Efficiently reduce the contamination levels of waste from ILW to LLW
- To Keep the reactor cavity water as much clean as possible to improve visibility and reduce source term for the tasks relating with Internals and vessel dismantling

Two processes were used to decontaminate the whole primary system (SG, PZR, RCP, loops, internals and vessel) and the other main systems (as RHR, CVCS, etc)

- DfD (Decontamination for Decommissioning)
- NITROX

Decontamination, final main data:

Total activity removed (Ci):	**802**
Total Co60 activity removed (Ci):	**714**
Metal removed (kg):	**234**
DF* - Steam Generator	**12**
DF* - Auxiliary Systems	**33**
DF* - Loops (primary circuit pipes)	**8**
DF* - Pressuriser	**50**
Spent resins (m3)	**13**

**DF: Decontamination factor achieved*

Collective doses (whole process) _ mSv-p	
Systems modifications	23
Maintenance / inspections	11
Decontamination process	25
Spent resins conditioning	60 (50%)
TOTAL DOSE (mSv-p)	119

The estimated dose for the te Whole decommissioning Project was stablished in 6.4 Sv-p. The dose “invested” for decontamination represents less than 2% of the decommissioning estimated dose.

The average área dose rate was reduced from 129 microSv/h, before decontamination, to 11 microSv/h after decontamination.

22 area dose rate measurement reference points were set in the Containment building to screen all the decommissioning work areas.

There were also fixed some points of contact dose rate measurement with some large equipments. In this case, the máximun reduction factor was obtained in contact with the Pressurizer, reaching a value of 50.

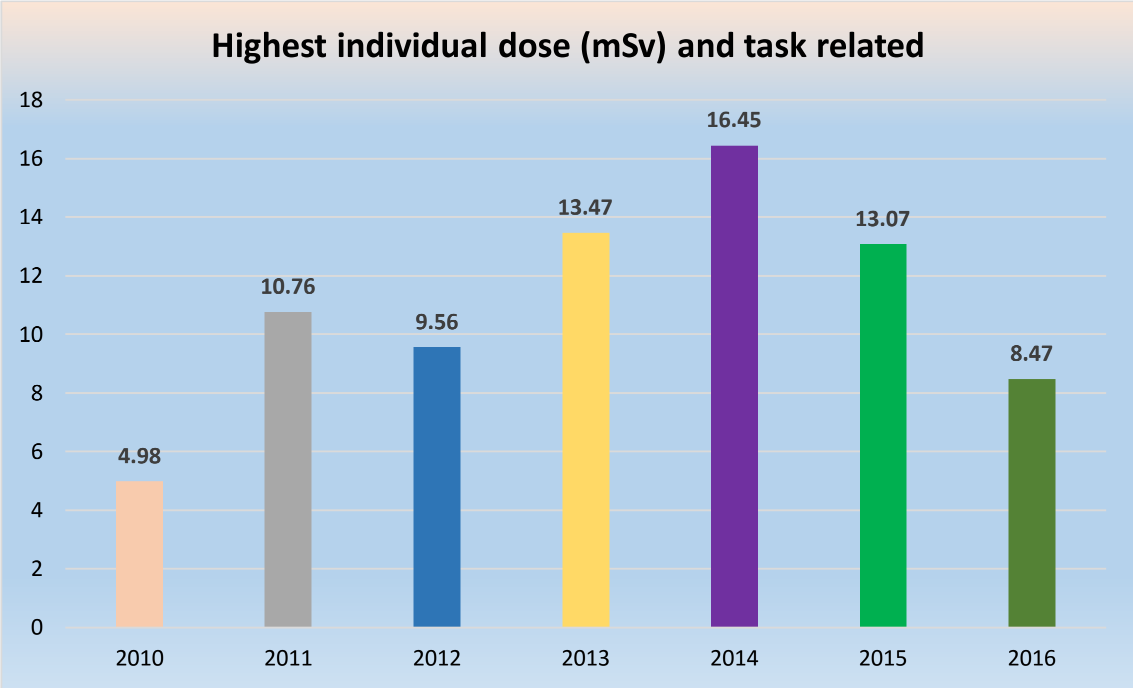
The estimated dose for the tasks in áreas affected by the decontamination process was: 4.1 Sv-p and the collective dose received in this tasks was 1,3 Sv-p, so the reduction in dose, attributable to the decontamination process is: 2.8 Sv-p

<u>Current situation of the decommissioning project</u>

- 75% project progress
- RCS and auxiliary systems segmented
- Removing the Biological Shield
- Removing the Cavity and Spent Fuel Pool contaminated walls
- Estimated completion date: December 31, 2018

The radiological risk has decreased substantially, as well as technological complexity.

Future works are focused on the decontamination of concretes, the demolition of buildings and the site restoration.



Collective dose per task, updated at March 31st, 2016

Cum.	
142.77	***RV internals***
234.35	***Reactor vessel***
329.71	***Steam generator***
16.64	***Pressurizer***
122.54	***Pump & legs***
846.00	*total*

Figures in mSv-p

WORK GROUPS	***Collective dose***	
Plant Modifications	*144.11*	*7%*
Maintenance & surveying	*347,03*	*16%*
Large components	*846,00*	*39%*
Decontamination in situ (components)	*29.03*	*1%*
Spent fuel pool conditioning & decon	*95.79*	*4%*
Dismantling in Containment building	*177.67*	*8%*
Dismantling in other buildings	*180.73*	*8%*
Biological shielding removal	*18.71*	*1%*
Contaminated concrete removal	*120.34*	*6%*
Surfaces decontamination	*20.81*	*1%*
Decontamination workshop	*0.25*	*0%*
Rad Waste managing	*181.61*	*8%*
total	***2162.07***	

POSTERS

Poster 1. Radiation Protection Studies related to the Ultimate Back-up Diesel Generator Installation

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After the earthquake and the following tsunami, the FUKUSHIMA nuclear plant lost all its electrical supply. To avoid this situation and to meet the French Nuclear Authorities technical prescriptions, EDF has put in place a very ambitious actions plan to further improve his nuclear plant safety. One of the technical answers is to install an Ultimate Back-up Diesel Generator on the whole French Fleet.

Even if the majority of works are planned outside controlled area in the UBD building (civil engineering, electromechanical works), Radiation Protection issues have to be taken into account in particular to:

- Evaluate the dose taken to proceed to the electrical connection between the UBD building to the unit (necessary works in controlled area via fuel building especially) and leading to forecasted collective dose between 10 and 40 man.mSv,
- Evaluate the impact of this modification on the zoning of the junction room outside the Fuel Building (due to a wall hole dedicated to pass the cables through),
- Evaluate the impact of the Effluent Treatment Building (at the vicinity of the UBD building) on dose rates on the outside work site (direct radiation and sky shine effect) in order to determine if specific working conditions dedicated to restricted areas have to be implemented.

For each case, specific 3D calculations have been performed with the PANTHERE code or MCNP code. The results of these RP studies will be detailed in the final paper.

Poster 2. Digital Spectrometry of Mixed Radiation Fields in Nuclear Facilities

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The newly developed two-parametric digital spectrometric system was used to measure of mixed neutron and photon fields around the IBA cyclotron located in the Positron Emission Tomography in Masaryk Memorial Cancer Institute in Brno, measurements of the mono-energetic neutrons generated by the Van De Graaff accelerator in the laboratories of Experimental and Applied Physics in Prague, measurements of the neutron and photon radiation in the active zone of the LR-0 research reactor in the Nuclear Research Institute in Řež and experimental measurements of the secondary neutrons in the irradiation room of Proton Therapy Center in Prague. The organic scintillation detector, types BC-501A, EJ-299-33 and Stilbene detector were utilized for the detection of neutron and photon radiation. Signals from the detectors are connected to digital spectrometer employing two analog-digital converters with very high sampling frequency (up to 2 GHz). The converters operate with a resolution of 12 bits.
Measured data are transferred to computer via XGMII (10 Gigabit Media Independent Interface) and identified by pulse shape discrimination (PSD) method. The quality of resolution PSD has been evaluated and the results are presented.

Poster 3. Gadolinium and Terbium-covered Bismuth-loaded Plastic Scintillators for Thermal Neutron Detection

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NEUTRON radiation detection forms a critical branch of nuclear-related issues, whether flow monitoring on industrial infrastructures, dose rate monitoring for radioprotection or radiological material detection addressing CBRN threats are concerned. The last decade has been driven by the quest for competitive alternative technologies to neutron detectors based on the helium-3 isotope whose announced worldwide shortage has generated massive market value fluctuations. In this context, the loading of plastic scintillators with organometallic complexes has shown a high potential for the deployment of sensitive and cost-effective detectors. Bismuth has thus been extensively studied as a dopant for low and medium-energy gamma-ray pseudo-spectroscopy. Now the implementation of gadolinium, which exhibits the largest cross section (48890 barns) available among stable elements for the radiative capture of incident thermal neutrons, as a converter used in conjunction with bismuth-loaded plastic scintillators opens an apparently straightforward potential for competitive innovation. A technical conundrum however lies within the separation of the scintillation signal due to the prompt gamma-rays most significantly emitted after the (n,γ) neutron radiative capture from the scintillation signal attributable to ambient gamma-rays and falling into the same energy range. To address this issue, the authors propose a neutron detection system based on three key features:

1) the use of natural gadolinium as a covering of the scintillator sensor to convert incident thermal neutron radiation into a photon (prompt gamma-rays and X-rays) and electron (internal conversion and Auger electrons) source term;
2) a two-identical bismuth-loaded plastic scintillator system, where:
   - the first scintillator, named *detection scintillator*, is covered with a gadolinium foil in which incident thermal neutron absorption occurs, while both the scintillator and the foil interact with incident photon radiation (and more marginally incident fast neutrons);
   - the second scintillator, named *compensation scintillator*, is covered with a terbium foil of same thickness as the gadolinium one. Both the scintillator and the foil interact with the photon and fast neutron part of the incident radiation, while thermal neutron absorption in terbium is negligible. In order to select energy windows of interest when studying the response to radiations, the sensors must allow for low-energy pseudo-spectroscopic performance through the enhancement of photoelectric and Compton interactions, hence the choice of bismuth-loading for the scintillators;
3) a unit for the compensation treatment of the signals collected on both channels, which isolates the signature of the interaction between thermal neutrons and gadolinium in the detection scintillator by subtracting the compensation scintillator response (incident photon and fast neutron radiation) to the total detection scintillator response (incident photon, fast and slow neutron radiation).

This paper describes the simulation-based scaling of the sensors and insulation of measurement channels, the synthesis of bismuth-loaded plastic scintillators, the experimental energy calibration of these scintillators, the nonlinear smoothing of the raw counting signals and the setting of a hypothesis test to discriminate the signal generated by the radiative signature of thermal neutron captures in gadolinium from statistical fluctuations over the compensation of both independent channels. The neutron measurement by compensation carried out over two gadolinium- and terbium-covered bismuth-loaded plastic scintillators has allowed the detection of a neutron activity in the photon radiation background of the californium-252 source at the exit of the HDPE block (estimated about 1.2 $\mu Sv.h^{-1}$ with a standard radiameter, which matches the typical constraints of a "monitored area" in the context of radioprotection applications), thus validating the concept of a new neutron detection system, robust to "controlled area" typical photon background, portable, compatible with online implementation, and whose cost-effectiveness advantageously compare to inorganic, silicon-based compensation schemes. In order to increase the counting rates, as well as the precision associated to them, future works will be turned towards a scale-up over the loaded scintillating samples, with all the known challenges associated to such a process: homogeneity issues with the loading and self-absorption of the scintillator to quote the more obvious of them.

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Poster 4. iPIX, a New Gamma Camera for Hot Spot Location

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We describe CANBERRA's new gamma imaging product, the iPIX, and related services. We discuss the features, functionality and performance of the iPIX. The iPIX makes use of coded aperture technology to provide quick localisation of gamma activity from a distance while estimating the dose rate at the measurement position. This tool is ideal for mapping a radioactive area prior to entry, reducing dose exposure for personnel during operations and operating to ALARA principles. The iPIX brings advantages over many other technologies with its compact dimensions, high portability, ease of operation and near-real time measurements, providing a highly flexible tool for gamma localisation.

We also discuss how the iPIX can be used in conjunction with other detectors in more complex situations where the location of the radioactivity, identification and quantification of nuclides and dose rate mapping may be required to support plant characterisation or ALARA studies. Examples of work carried out by the iPIX and results, including located hot spots, are presented.

Poster 5. Combined uncertainty for nuclide specific free release measurements – a practical method

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A method has been developed for the calculation of combined uncertainty for nuclide specific free release measurements by gamma spectrometry at the Forsmark NPP. The purpose is to show that the free release limits given by the Swedish Radiation Safety Authority (SSM) are met at a certain confidence level.

The free release criterion for a case where several nuclides are present in an item, consists of a sum value that should be less than or equal to one. The sum is calculated by adding the amount of each present nuclide divided by its nuclide specific free release limit. A model equation has been defined to determine this sum value. The components that are included in the equation are: the measured Co-60 radioactivity, a factor that takes into account the free release limits and the chosen nuclide vector for the measurement, and also correction factors for the measurement efficiency model, the homogeneity of the measured item and deviations from the assumed material composition of the item. Each component of the model equation has been assigned a standard uncertainty, and uncertainty propagation has given the combined uncertainty.

To allow the combined uncertainty to be easily determined by the measurement system operator after each measurement, an expression has been given for the combined uncertainty as a function of the uncertainty contribution that is reported directly from the gamma spectrometry system. This includes uncertainties that arise from event counting, branching ratio and decay correction.

For an estimated 95 % confidence level, the calculated sum value (according to the model equation) plus 1.645 times the combined uncertainty should not exceed the free release condition.

The estimation of the combined uncertainty follows the GUM (Guide to the expression of uncertainty in measurement) methodology as described in ISO/IEC Guide 98-3:2008. Simplifications have been done to the uncertainty propagation calculation, so that a Microsoft Excel spreadsheet, a so called Kragten Spreadsheet, can be used to estimate the combined uncertainty (Kragten, Analyst, October 1994, Vol. 119). The spreadsheet also shows which component uncertainties that contribute most to the combined uncertainty. This is important information for decision making regarding method improvement. Improvement is encouraged by the use of a confidence level in the free release process, since less uncertainty most likely makes it possible to approve more material for free release.

Poster 6. New Sensor Design for Monitoring a Radioactive Effluent

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Due to current environmental events, radioactive effluents represent a major topic. Therefore, the need for an effective effluent management system is increasing and represents a big challenge for nuclear power plants. For categorizing effluent especially in case of a discharge optimization, the real time detection is crucial. In the case of α & β monitoring, no reliable tools for real-time detection are available.

The proposed approach answers to real time α, β monitoring issue. It consists in using a new sensor based on scintillating optical fibers for optimized α & β detection. The system performs a continuous monitoring of radioactive effluents and has the ability to provide an early alarm in case of a radioactive contamination depending on the alarm threshold.

This paper presents the sensor developed and the results obtained in our laboratory, experiments were carried out in the framework of the European project SAFEWATER - Innovative tools for the detection and mitigation of CBRN related contamination events of drinking water.

Poster 7. False Alarm Reduction in Portal Monitors utilizing the FastTrack-Technology

Tobias Baer[1], Daan van Bree and Christian Günther
for the Mirion Technologies Team

Mirion Technologies

Many colleagues at Mirion Technologies have been involved in the development of the FastTrack-Technology and its implementation in various products. Their contribution is greatly appreciated.

Introduction

Portal monitors to scan vehicles and pedestrians for ionizing gamma radiation are widely used, e.g. in the nuclear industry, in homeland/event security applications or in radiological emergency scenarios. Unfortunately, a lot of false alarms are occurring due to the fact that commonplace portal monitors are not able to distinguish between sources being carried through the monitor and sources that are located nearby but outside the pillars of portal monitors. Due to the nature of a single detector any ionizing radiation - irrespective of the location from which it irradiates - generates a measurement effect. Significant false alarm rates are the consequence leading to a lack of confidence in the used technology, especially in situations where a lot of persons want to leave the area but need to wait for clarification of the alarm results.

In the following, a novel method - Mirion Technologies’ patented FastTrack-Technology - is presented which, amongst other positive features, is able to clearly distinguish between sources being carried through the portal monitor´s pillars and those sources or contaminations being located nearby but outside the monitor. The FastTrack-Technology is therefore particularly useful in challenging background conditions, e.g. at Chernobyl or Fukushima but also for applications in which high frequency measurements are expected. This may involve outages of NPPs, radiological terror prevention at airports/customs or large scale events/venues. After having explained the basics of FastTrack-Technology, examples of real life applications are presented.

Basics of FastTrack-Technology

The basic idea of the FastTrack-Technology is both very simple and very effective: In contrast to commonplace portal monitors which are composed of a single detector per pillar, the FastTrack-Technology is based on three (or more) horizontally arranged detectors per pillar as illustrated in
figure 1.

Depending on the direction in which a source is moving through the monitor, the detector electronics respond in a different way: given a source approaches from the left, the “black” detector responds first whereas the “green” detector responds last. Furthermore, the sequence of the signals peak amplitudes in this example is black → red → green. A source approaching from the right would reverse the sequence into green → red → black. Any movement in between these two initial positions would cause a smooth transition in between the signal sequences and amplitudes.

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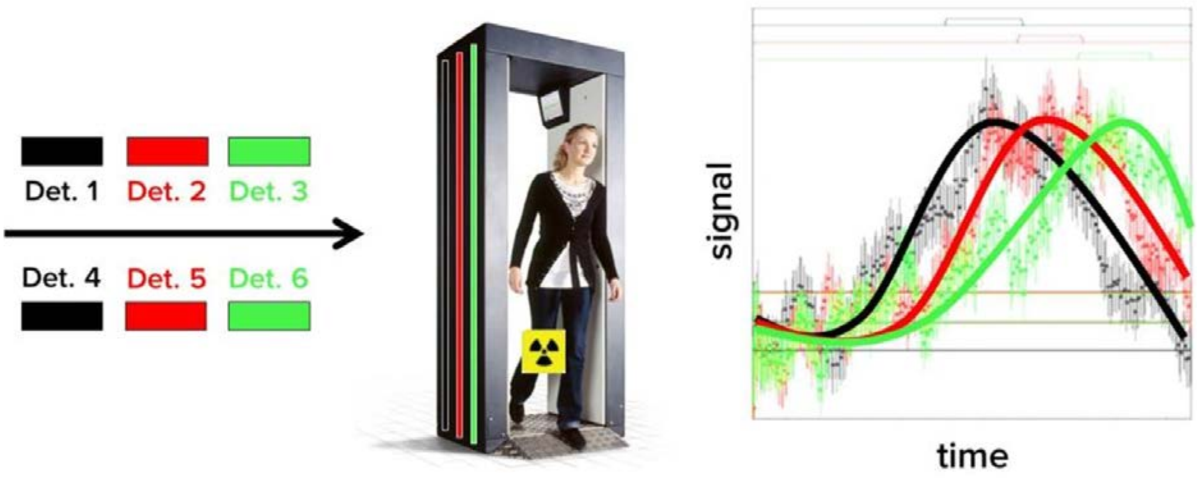

Figure 1: Detector arrangement in FastTrack-Technology; example: Mirion Technologies' FastTrack-Fibre™

As shown in this simplified example, the direction of movement of the source can be determined based on the temporal profile of the measurement signals from all three detector arrays. Mirion Technologies' FastTrack-Technology compares this information to the direction and speed of movement of the object or person currently being measured (which is independently determined by multiple light barriers). This allows a reliable distinction of sources being carried inside/through the monitor from sources and dynamic background fluctuations originating from outside the monitor.

Limitations to this approach are mainly given by electronics noise, geometrical imperfections, and complex source geometries. Thus, a down to the "degree" resolution of the source's direction of movement is not possible today.

Successful measurement operation of the FastTrack-Technology is not limited to sources or contamination being moved but can also handle staircase signal shapes which e.g. occur when doors to a close-by radiologically controlled area are opened. In this case all detectors would respond with an identically temporal staircase signal profile which can be clearly distinguished from the specific profile of a source/contamination being carried through the monitor (sinusoidal shape).

In this way the main reason of false alarms in portal monitors is eliminated both for moving sources as well as static ones.

Furthermore, the FastTrack-Technology ensures, thanks to its advanced algorithms, measurements without waiting times (walk-through mode) of pedestrians or vehicles without compromising MDA-levels.

Examples

Figures 2-4 are showing typical measuring situations in which commonplace portal monitors would respond with false alarms whereas FastTrack-Technology ensures trouble free operation:

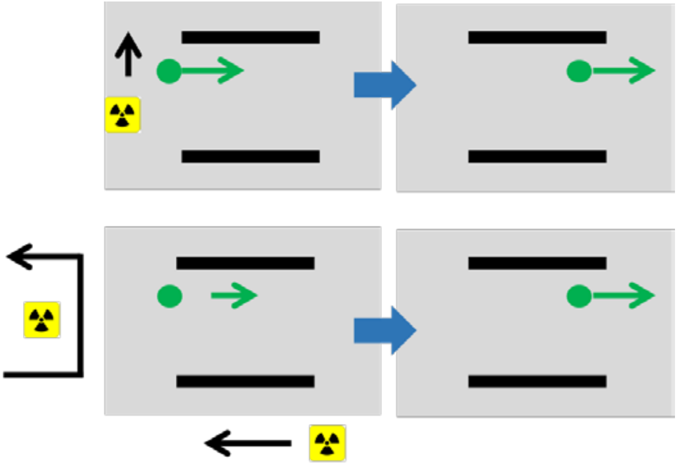

Figure 2: Moving sources during successful operation of Mirion Technologies' FastTrack-Fibre™

While sources are present in the vicinity of a FastTrack monitor during measurement operation the signals of all detectors are significantly different from each other, both in chronological order as well as in amplitudes. As explained above, the FastTrack algorithms can reliably distinguish the fraction of the signal originating "inside" the monitor from the contributions from outside sources. For the sake of precaution, the FastTrack monitors can be configured to highlight measurements where external effects are detected by descriptive indications such as "external contamination" or "non-dynamic" in order to provide additional information to the radiation protection engineer.

In addition, the FastTrack-Technology prevents an (un-)intentional extraction of sources from controlled areas, as depicted in figures 3 and 4: when a source is carried into the monitor and remains inside, the rising edge of the detectors signals will not proceed to a trailing edge when the person leaves the monitor without the source. A corresponding alarm is given by FastTrack monitor.

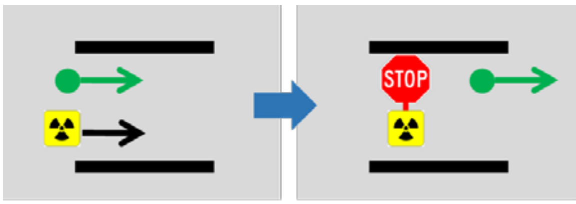

Figure 3: Source remaining inside the monitor

The same is done when a stationary source inside the monitor is extracted from the monitor. In this case, a trailing edge of the detectors signals is detected without rising edge in the beginning of the measurement. The FastTrack monitor will give a corresponding alarm.

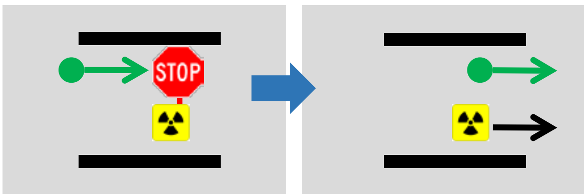

Figure 4: Source taken away from inside the monitor

Notwithstanding the examples above, all "true" contaminations are detected by the FastTrack-Technology – irrespective of sources inside or outside the monitor.

FastTrack-Vehicle™ monitor in Fukushima

Mirion Technologies' patented FastTrack-Technology has been invited for a test of vehicle monitors nearby Fukushima, Japan for demonstrating its efficiency. Measurements of all vehicles leaving the 30km area around Fukushima have been performed with an MDA of 4 Bq/cm^2 Cs-134 at walking speed. Mirion Technologies' FastTrack-Vehicle™ monitor has been chosen for this test. The two pillars where placed at a distance of 6m clear width, as illustrated in figure 5.

Figure 5: FastTrack-Vehicle™ in operation nearby Fukushima

For the project team it was of particular interest how the FastTrack-Technology would perform in high backgrounds of about 1–2 µSv/h due to contaminated soil. All measurements had been carried out over a period of three weeks. In the very beginning, additional steel plates had been mounted above the soil to artificially decrease the background, see figures 6 and 7. However, this countermeasure proved to be unnecessary as the FastTrack-Vehicle™ could fulfil all requirements without these additional means.

Figure 6 and 7: FastTrack-Vehicle™ in operation nearby Fukushima

FastTrack-Fibre™ pedestrian monitor at major sporting event

As a counter terror measure during a major sporting event it was of highest importance to scan all (100%) of the approx. 4.1 million spectators, attendees, politicians, security personnel, etc. for radiological threats when entering the arenas. Due to the expected large number of accesses at each access point the organizers expected an uninterrupted passage of pedestrians into the arenas, i.e. no tailbacks were allowed, but people should be able to move more or less "as they like". As a solution to these challenging requirements each access point was equipped with a dedicated FastTrack-Fibre™ portal monitor, as shown in figure 8.

Figure 8: Mirion Technologies' FastTrack-Fibre™ pedestrian monitor

Entering the portal monitors was only possible through restricted guidance systems, as illustrated in figures 9 - 11.

Figure 9: Schematics of arrangement to guide visitors incl. radiation scan by FastTrack-Fibre™

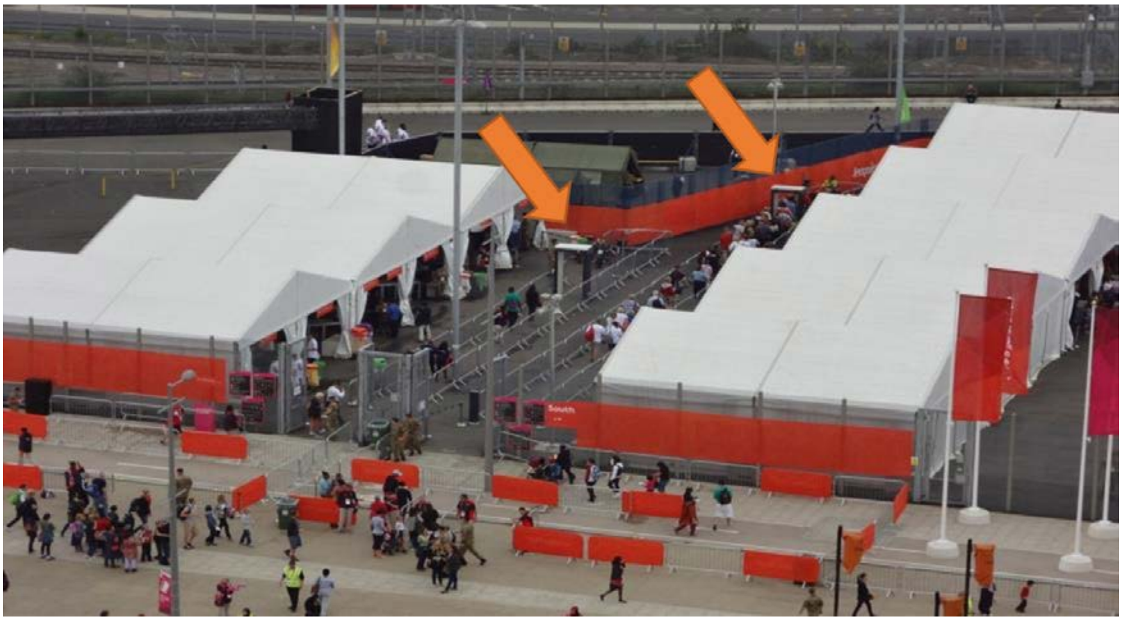

Figure 10: Live situation of visitors entering arenas incl. radiation scan by FastTrack-Fibre™

Figure 11: Live situation of visitors entering arenas incl. radiation scan by FastTrack-Fibre™

When a contamination has been detected, a group of 1 - 3 persons was isolated from the crowd and each person of the group was manually checked with handheld spectroscopic devices.

This measurement campaign for radiological threats has been the first of its kind and dimension, worldwide. Some impressive figures have been reported:

- 4.1 million measurements have been executed by 19 FastTrack-Fibre™ monitors over a period of 19 days.
- This corresponds to over 10.000 measurements per day and monitor.
- The most intensively used monitor carried out over 300.000 measurements within 19 days.
- Throughout the whole period 72 alarms have been detected. All resulting from medical treatments of the corresponding persons.
- No false alarm events were generated in the 4.1 million measurements.
- Due to these impressive figures the CBRN "Counter Terror Award" has been awarded to the FastTrack-Fibre™ monitors.

Conclusion

The FastTrack Technology has been invented to reduce false alarms in portal monitoring for ionizing gamma radiation. It is based on a sequential three-detector-arrangement, which allows correlating the measurement signals to the movement of the measurement object (or person). Consequently, it can be clearly distinguished if a source of ionizing radiation is moving with the measurement object through the monitor or if it is located outside the monitor. This leads to a significant false alarm reduction w.r.t conventional portal monitors, especially in challenging background conditions.

Two applications have been presented:

- Mirion Technologies' FastTrack-Vehicle™ monitor has been operated near Fukushima. The test demonstrated a robust performance also in significantly elevated gamma background of 1–2 µSv/h.
- Mirion Technologies' FastTrack-Fibre™ pedestrian monitor has been used for a large-scale radiological screening at a major sports event. Throughout the event 4.1 million measurements have been performed by 19 monitors. Not a single false alarm occurred.

FastTrack Technology has been developed and patented by Mirion Technologies (RADOS) GmbH.
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Poster 8. An Innovative Isotope Identification Approach for Plastic Scintillator

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Radiation Portal Monitors (RPMs) are usually used in nuclear power plants to detect radioactive sources and especially gamma sources transported in vehicles or carried by pedestrians. These RPMs represent the ultimate barrier between public area and controlled area. Generally composed by several large plastic scintillators, these kind of system has a limited identification ability. The isotope identification is generally done from spectra with higher intrinsic resolution detectors such as germanium. The non-loaded plastic scintillators are generally unfit to perform isotope identification due to the lack of photopics which induces a weak energy resolution. However, several techniques have been developed to solve this issue. The most widely used among them divides the spectrum into multiple windows of interest [1]. This technique is commonly known as "energy windowing" but is relatively limited and does not allow to discriminate sources containing components in high and low energies like NORM (Naturally occurring radioactive material).

The proposed approach [2] is a new isotope identification principle to deal with the overall range of nuclear detectors. This method allows the identification of isotopes using detectors with poor intrinsic resolution and lower counting rates than the existing techniques. It’s performed on a distance, from current measured isotope against a set of isotope references stored in a database.

The solution presented in this paper allows to add an identification function to many existing systems performing a simple detection function without needing major hardware changes.

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Poster 9. Internal Dosimetry of Gamma Emitting Radionuclides at CNE Cernavoda

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ABSTRACT

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

Two internal contamination events are presented in this paper, one by inhalation of ^{95}Zr -^{95}Nb and the other one by percutaneous transfer of ^{137}Cs. IMBA Professional Plus code is used in present to calculate internal doses due to gamma emitting nuclides intakes. Dose calculation results are presented for both events.

1.0 INTRODUCTION

Cernavoda NPP has two CANDU 6 reactors in commercial operation, first since December 1996 and the second one since November 2007.

For a CANDU 6 type reactor the major contributor to the external dose is gamma radiation (about 90%). Tritiated heavy water (DTO) is the major contributor to the internal dose of professionally exposed workers contributing with up to 40% of the total effective collective dose.

The main purpose of Individual Dosimetry Program design and implementation is monitoring, evaluation and recording all the significant individual radiation doses received by an individual during activities performed at Cernavoda NPP from both external, Hp(10), and Hp(0.07) and internal, E50, exposures and providing support to maintain ALARA all radiation exposures.

Health Physics Department provides individual dosimetric surveillance for both external and internal exposure for all the personnel entering radiological controlled areas: Cernavoda NPP employees, contractors, short-term atomic radiation workers, and visitors. Individual dose monitoring is provided by our dosimetry laboratory licensed by Romanian regulatory body, National Commission for Nuclear Activities Control (CNCAN).

Radiation Protection Program for Cernavoda NPP is designed insomuch as legal dose limits, established by our national regulatory body, for atomic radiation worker and for public, will not be exceeded.

Cernavoda NPP Dose Monitoring, Evaluation and Recording program is based on the latest ICRP recommendations: ICRP publications 75, 60, 66, 68, 78, 103 and also, on the requirements of national laws and regulations.

2.0 INTERNAL DOSIMETRY PROGRAM

CANDU is a Canadian design pressurized heavy water reactor - PHWR, which uses natural uranium fuel and heavy water as a neutron moderator and primary heat transport agent.

Activation of deuterium by thermal neutron flux is the major mechanism of tritium production in a PHWR reactor but other nuclear reactions could also produce tritium (ternary fission, re-conversion of ^{3}He to ^{3}H by thermal neutron reaction).

Tritiated heavy water (DTO) which is present at many work locations as water vapors is the major contributor to internal dose of professionally exposed workers at Cernavoda NPP.

Minor contributors to the internal doses could be:

1) activation products as ^{95}Nb and ^{95}Zr (corrosion products from the zirconium alloys in fuel sheath and pressure tubes) and ^{60}Co (corrosion product from steel alloys reaching the active zone). Other activation products which could be present in PHT are ^{51}Cr, ^{54}Mn, ^{59}Fe, ^{113}Sn, ^{124}Sb.
2) fission products as ^{131}I, ^{133}I, ^{135}I and ^{137}Cs.

Radiation protection programs provide adequate basis theoretical, and instrumental for exposure control designed to minimize the combined external and internal dose i.e., total effective dose equivalent. As related to internal dosimetry, the radiation protection program includes the following elements: job specific / area scaling factor, estimates of intake and assessment of dose. Biological retention and excretion models are provided by internal dosimetry software – LUDEP 2.0 or recently IMBA.

Dose assessment from bioassay generally relies firstly on intake calculation for a radionuclide either from direct measurements with a Whole Body Counter – WBC or by indirect measurements, e.g. urine, faeces or working area contamination samples. Predicted values of the internal residual quantities for unit intake are estimated by using ICRP biokinetic models and these values are used to estimate the intake (ICRP, 1997b). The effective dose resulting from a particular intake is then calculated using ICRP dose coefficients.

2.1 Internal dosimetry for minor contributors

The most frequent source for internal contamination is due to the presence of loose contamination with ^{95}Nb / ^{95}Zr, when primary heat transport system is opened for maintenance activities.

Routine WBC monitoring program is implemented as follows:

- The fuel handling / fueling machine personnel are monitored once a month, and the rest of Fuel Handling staff four times a year.
- Nuclear operators, maintenance services workers, radiation control service technicians, chemical lab technicians and Non Destructive Examination - NDE laboratory technicians are monitored once a year.
- Employees which usually are not involved in activities in contaminated areas are monitored every three years. New employees are monitored at the very beginning of activity in the radiological zone.

In case of a job with significant risk of internal contamination, every worker involved is monitored at the Whole Body Counter prior the beginning of work and immediately after the end of that special activity. WBC monitoring is also required after a personal contamination event (skin or clothes).

Calculation method of committed dose E_{50} due to intake of beta – gamma emitting particulates is based on ICRP recommendations.

2.2 Instrumentation

Internal contamination with gamma emitting nuclides at Cernavoda NPP is measured by using the Whole Body Counters - WBC Fastscan, and Accuscan II Canberra USA.

FASTSCAN whole body counter is designed to quickly and accurately monitor people for internal contamination of radionuclides with energies between 300 keV to 2000 keV. The system includes two large sodium iodine detectors [NaI(Tl)] that typically provide a prior Lower Limit of Detection of approximately 150 Bq for ^{60}Co with a count time of one minute for a person with normal ^{40}K internal activity. This dual detector system provides uniform or flat (±15%) response along the longitudinal axis from the thyroid of the tallest 99th percentile male to the lower gastrointestinal tract of the shortest female. For a counting time of 3-5 minute range, the detection limits are about 50 Bq for ^{137}Cs and about 60 Bq for ^{60}Co.

The ACCUSCAN-II is a high resolution, stand-up WBC. It is designed to identify and quantify radionuclides with energies between 100 keV and 2000 keV in complicated combinations. The ACCUSCAN-II also provides information on the location of the radioactive materials found in the body through its scanning mechanism and the system's ABACOS software. Count time is the 3-5 minute range. The detection limit for uncontaminated person (count time 30 minute) is about 81 Bq for Cs-137 and about 76 Bq for Co-60.

2.3 Estimates of Intake and Assessment of Dose

Investigative whole body counts provide a timely and direct measure of the deposition of gamma emitting radionuclides in the body. An estimate of the intake of gamma emitting radioactivity is determined from these whole body count measurements and applicable intake retention fractions.

$$I = \frac{A(t)}{IRF(t)}$$

where:

I – estimate of intake in Bq

A(t) - the bioassay measurement obtained at time t in Bq

IRF(t) - intake retention fraction corresponding to the nuclide, solubility class, mode of intake, time after intake, and type of bioassay measurement

The committed effective dose (CED) from gamma emitters can be calculated using the estimates of the intake (I) and the Dose coefficients in the ICRP:

$$CED = e_{50} \times I \ (mSv)$$

where e_{50} is the dose coefficient, i.e. the committed effective dose per unit acute intake calculated for 50 years, for adults in Sv/Bq.

2.4 Work control

Routine monitoring of workplaces is intended to demonstrate that the working conditions allow continuing the activities and there have not been changes that compel modification of the working procedures or protective measures.

Routine program includes monitoring of:
- beta, gamma and neutrons dose rates;
- tritium in air concentration;
- aerosols (alpha, beta, gamma);
- iodine in air concentration;
- surface contamination level.

An electronic database HAZARDINFO is available for plant staff, RP technicians and workers, containing information about all significant radiological hazards in accessible areas.
For particular jobs with known contamination risk preventive actions intended to prevent the spread of contamination on the surrounding areas and internal contamination of workers are mentioned in Radiation Work Permit: set-up of rubber areas / rubber change areas, installing ventilated tents with HEPA filers, using adequate individual respiratory protection equipment, continuous monitoring of air borne radioactivity.

Various respiratory protection equipment are available for workers performing radioactive contamination risk jobs depending on working area particular conditions, 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 due to unexpected airborne contamination or unintended contact with contaminated equipment. The most common route of entry of a radioactive contamination into the body of a worker is by inhalation of airborne radioactivity.

After few years of increasing number of internal contaminations, most of them with doses below the recording level of 0.1 mSv/month, a root cause analysis performed in 2010 identified two main causes for these contaminations: spreading of radioactive particles from contaminated protective equipment (tyvek hoods) during undressing, followed by inhalation, and temporary removal of individual respiratory protection equipment (air supplied / particulate filter half mask) in order to communicate with co-workers, followed by putting the mask back on the mouth and nose.

Procedures for undressing tyvek hoods used during activities with high risk of contamination were improved with the requirement provide assistance to workers during undressing protective clothes.

Wireless ear-phones were purchased in order to allow communication without removing the mask.

A strict control of contamination events allow to promptly identify gamma radionuclide intakes and initiating bioassay procedures for internal dose calculation. Since the moment of intake is very important for an accurate calculation of internal doses, when the risk of contamination is identified prior performing the job, workers must be monitored at whole body counter before and immediately after the work. WBC requirement is documented in the Radiation Work Permit, and the workers are informed during pre-job briefing.

3.0 RESULTS AND DISCUSSIONS

Despite a careful planning for radioactive source control, the spread of radioactive contamination in working environment could lead, infrequently, to radioactive material intake by workers.

Most of them involved activities of the radionuclide close to the MDA of the WBC and probably are external contamination or particles retained in the superior air ways, taking into consideration that the activity is not identified on measurements performed after taking a shower or blowing the nose, in the same day or in the next morning.

Two confirmed internal contamination events are presented in this paper, one by inhalation of ^{95}Zr -^{95}Nb and the other one by percutaneous transfer of ^{137}Cs. IMBA Professional Plus is used in present to calculate internal doses due to gamma emitting nuclides intakes. Dose calculation results are presented for both events.

3.1 Inhalation of ^{95}Zr -^{95}Nb

During Unit 2 planned outage, on May 21st 2015 a welder had to replace a defective segment of a tubing line on a Primary Heat Transport system auxiliary. Prior the welding activity the pipe needed to be dried by blowing with argon and an argon shield had to be maintained in the pipe during the welding. Since the activity was performed in a narrow space and tritium concentration was low, the welder didn't wear plastic suit. The operating experience for similar jobs did not reveal the hazard of airborne contamination spreading during this step of the job, the worker did not wear other respiratory protection. While exiting reactor building he was found contaminated and internal contamination was confirmed at the full body counter.

Whole body counting was performed by using NaI(Tl) detectors FASTSCAN system designed to quickly monitor people for internal contamination of radionuclides with energies between 300 keV to 2000 keV. ^{95}Zr and ^{95}Nb are a parent-daughter pair in a transient equilibrium. The two radioisotopes have gamma radiation with energy 724 and 758 keV for ^{95}Zr respectively 765 keV for ^{95}Nb which are not resolved by NaI(Tl) detectors. A contamination of 18 kBq ^{95}Nb was initially measured. After removing external contamination by repeated showers the remaining activity of 7300Bq was measured, the first day after intake, an 1921 Bq in the second day which was taken into consideration to start calculating the internal dose.

Contamination samples from surfaces of equipment in working area were measured by gamma ray spectrometry. ^{95}Zr and ^{95}Nb were identified as main contributors, but other radionuclides such as ^{51}Cr, ^{54}Mn, ^{59}Fe, ^{60}Co, ^{113}Sn, ^{124}Sb having activities lower with several order of magnitude were identified as shown in the Table 1, below.

Table 1 Radionuclides in contamination sample from PHT system tubing replacement in Unit #2 planned outage in 2015 May 21st

Radionuclide	Activity (Bq)	Relative uncertainty	Ratio to ^{95}Nb activity
^{51}Cr	1.98E+01	25.45%	3.57E-02
^{54}Mn	1.25E+01	5.76%	2.24E-02
^{59}Fe	2.23E+01	6.07%	4.02E-02
^{60}Co	2.09E+01	4.12%	3.76E-02
^{95}Nb	5.56E+02	3.18%	1.00E+00
^{95}Zr	2.86E+02	2.48%	5.15E-01
^{113}Sn	5.41E+00	20.95%	9.74E-03
^{124}Sb	1.92E+00	25.33%	3.45E-03

In January 12th 2016 he was found with 180 Bq Nb^{95} on **FASTSCAN** whole body counter, and no contamination was found on January 25th.

An effective committed dose of 0.04 mSv was calculated by: IMBA Professional Plus, Issued in Dec 13, Version number 4.1.47, based on following parameters:
Initial intake: 14402 Bq of indicator nuclide Nb^{95} and 34% Zr^{95} associated nuclide;
Intake regime: acute inhalation
Absorption to blood: Type S (Slow)
Aerosol/deposition parameters were: ICRP Defaults
AMAD = 5 µm
GSD = 2.4977233
Density = 3g/cc
Shape Factor = 1.5
Worker Type = light
Total effective dose from indicator nuclide: 0.0207 mSv
Total effective dose from associated nuclide: 0.0189 mSv

3.2 Percutaneous transfer of Cs^{137}

On January 23rd 2015, a fuel handling operator had been contaminated on the skin during preventive maintenance of spent fuel handling tools in Unit #1. Gamma spectrometry performed on liquid waste resulted from handling tools decontamination identified other radionuclides besides ^{137}Cs (as shown in Table 2 below, but the whole body counting performed after the contamination event identified ^{137}Cs only.

Table 2 Radionuclides in fuel handling tools decontamination liquid waste sample, Unit #1 in 2015 January 26st

Radionuclide	Activity (Bq)	Relative uncertainty	Ratio to ^{95}Nb activity
^{54}Mn	1.45E+03	6.18E-02	2.70E-02
^{59}Fe	1.22E+02	3.26E-01	2.27E-03
^{60}Co	2.58E+03	3.86E-02	4.80E-02
^{95}Nb	5.38E+04	3.12E-02	1.00E+00
^{95}Zr	2.43E+04	3.68E-02	4.52E-01
^{137}Cs	8.37E+01	1.16E+00	1.56E-03
^{144}Ce	8.44E+02	6.09E-01	1.57E-02

Since no radioactive aerosols were identified in working area during and after the job by continuous air monitor - CAM, and also, no contamination was found on operator face, around the nose and/or mouth, the only hypothesis remained that the intake of Cs was only possible by absorption through the intact skin. Percutaneous absorption of cesium has been demonstrated in 1 and 2 hr in vivo exposures of rats. In 2 hr experiments the blood concentration of cesium reached its maximum 38–45 min after its application. [Richard H. Guy et al., 1999]

IMBA do not calculate internal doses based on absorption through skin and bioassay data after intake fits inhalation / ingestion and / or injection model. Cesium is readily and almost quantitatively absorbed in the GI tract and widely distributed throughout the human body, mainly in the soft tissues. Excretion appears to be bimodal: an average 10% of a single oral dose is excreted within 1–2 days, but the major part has a half-life of 50–150 days. ^{137}Cs was measured to have a physical half-life of 30 years and a biological half-life (the retention half-time corrected for radioactive decay) of 102 days. The distribution of radiocesium throughout the body and the β and γ radiation from its decay result in essentially whole-body irradiation. [ICRP Publication 78]

On September 16th 2015 the fuel handling operator was found with a ^{137}Cs body burden of 265 Bq on **FASTSCAN** whole body counter, and no internal contamination higher than the detection level of 50 Bq was found on October 7th.

An effective committed dose of 0.089 mSv was calculated with IMBA Professional Plus, Version number 4.1.47, based on following parameters:
Initial intake: 12893 Bq of indicator nuclide ^{137}Cs and no associated nuclide;
Intake regime: acute inhalation
Absorption to blood: Type M (Medium)
Aerosol/deposition parameters were: ICRP Defaults
AMAD = 5 µm
GSD = 2.4977233
Density = 3g/cc
Shape Factor = 1.5
Worker Type = light

The figure below presents the measured and calculated by IMBA whole body activity of ^{137}Cs.

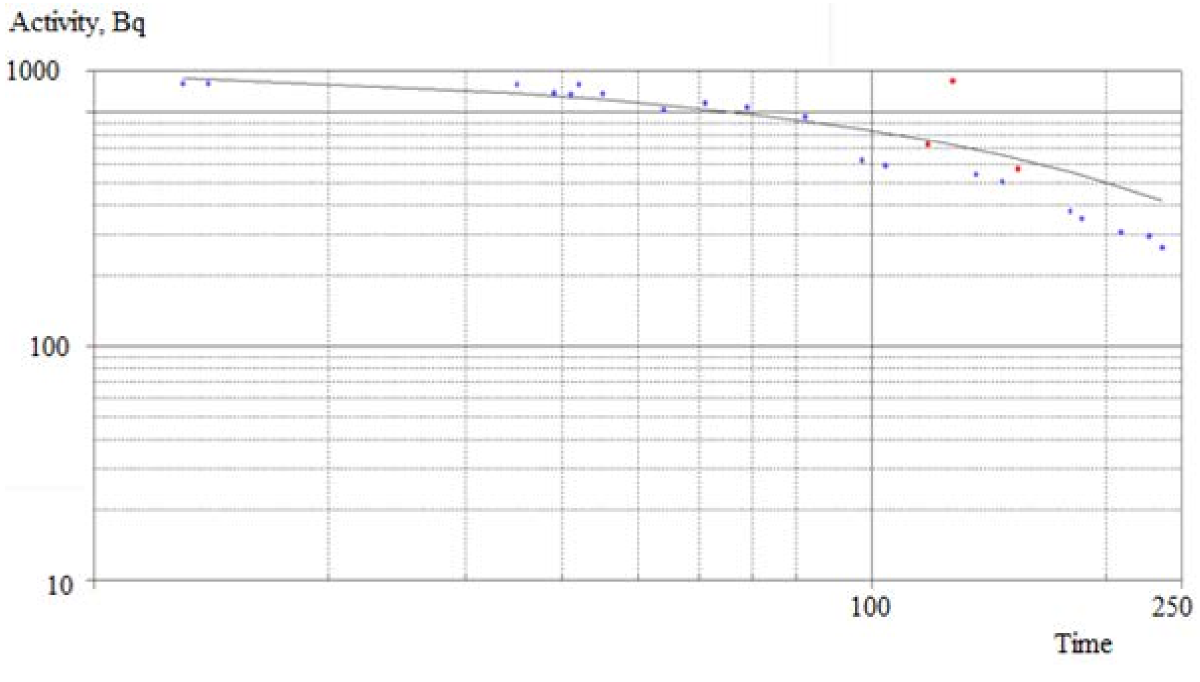

Figure 1. ^{137}Cs measured whole body activity and calculated by IMBA

4.0 CONCLUSIONS

Internal contamination hazard must be carefully controlled during maintenance activities which could spread radioactive contamination in working area. Individual monitoring for intakes of radionuclides for professional exposed workers is critical to provide valuable information to be used with biokinetic models in order obtain an assessment of the committed effective dose. This is necessary to demonstrate compliance with managerial and regulatory requirements and to contribute to the contamination hazard control during maintenance and operation of the plant ant to contribute to design improvement.

At Cernavoda NPP the operational radiation protection programs correlates assessment of workplace conditions, contamination control measures and individual internal exposures monitoring in a way that allows us to meet these objectives and, first of all to prevent as much as possible internal contaminations with beta / gamma and alpha radionuclides. Since 1996, when operation of Unit 1 started till 2013 most of internal exposures excepting tritium, were below the recording level of 0.1 mSv and, only very few internal contaminations led to doses above this value. Since 2014 till now no internal contamination led to recordable doses.

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Poster 10. Radiation Protection Education and Training Activities by the Belgian SCK•CEN Academy

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Since the early discoveries of ionizing radiation, a deep insight into the risks and benefits of applying radioactivity in daily commercial and research practice has been build up. The scientific world of radiological protection is in constant motion, triggered by new research as well as by developments and events in the daily industrial and medical sector. In addition, national and international standards, regulations and guidelines aim at steering daily practice and procedures that guarantee the protection of workers, the public and the environment.

With the SCK•CEN Academy, we transfer the latest insights in radiation protection to professionals dealing with ionizing radiation and to students. This presentation describes our activities in:

1. Guidance to students and young scientists in the domain of radiation protection:

The SCK•CEN Academy provides opportunities for Bachelor and Master students, PhD candidates and any professional interested in enriching his or her radiation protection competences. Teaching and research supervision are provided by members of the SCK•CEN research team. They share their knowledge 'from the frontiers of nuclear science' and oversee practical exercises that can be carried out using our centre's nuclear facilities. Final-year pupils and teachers can also discover the world of radiation protection via monthly visits to the radiation protection research laboratories.

2. Organisation of courses related to radiation protection:

The SCK•CEN Academy contributes to academic learning through collaboration with all Belgian universities and several universities abroad. For example, the Radiation Protection Expert (RPE) course is a one-year post-graduate course (20 ECTS) developed in line with the legal requirements for RPEs, as set in the Belgian royal Decree of 2001. It is targeted towards those who need to be formally recognised as RPE, as well as to all professionals working in nuclear, radiology or the medical sector.

Next to academic learning, the SCK•CEN Academy also provides customized training courses for professionals. For example, the course "Information and training in radiation protection for radiation workers" is aimed at workers who are possibly exposed to ionizing radiation in their professional environment, according to Article 25 of the Royal Decree of 20 July 2001. In addition, we also provide for continuous professional development (CPD) sessions in radiation protection for RPEs working in nuclear industry or in the medical sector, where we aim at discussing the most recent status of relevant subjects in radiation protection for professionals responsible for the supervision of exposed workers, public and environment. But also more specialised courses in dosimetry, emergency preparedness, radiation biology, etc. are provided.

3. Policy support in radiation protection education and training matters:

The implementation of a coherent approach to education and training in nuclear science and technology becomes crucial in a world of dynamic markets and increasing workers' mobility.

Through networking and participation in international programmes, the SCK•CEN Academy contributes to a better harmonisation of education, training practice and skills recognition on a national and international level. In the domain of radiation protection we coordinate the ENETRAP series of projects (6FP and 7FP), we participate to the IAEA steering committee for education and training in radiation protection, transport and waste safety and ad-hoc working groups of the OECD. We also organize the series of ETRAP conferences and are active in the Board of EUTERP.

4. Caring for critical-intellectual capacities for society:

Nuclear technology attracts attention and strong opinions in society. While the science itself may not be controversial, its application often is. Working with nuclear technology, either as scientist, manager or regulator, requires both technical knowledge and an insight in the societal issues. There is a growing awareness of the importance of being able to consider this wider context. The SCK•CEN Academy is unique in addressing this challenge by developing educational content and methods to raise awareness and stimulate thinking and discussion.

Poster 11. Reactor Pressure Vessel Closure Head Replacement of the Belgian Tihange 3 and Doel 4 units – Follow-up and On Site Dosimetry

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1. *Introduction*

According to a technical report from the IAEA referenced in [1], alloy 600 components such as control rod drive mechanism (CRDM) nozzles have experienced primary water stress corrosion cracking (PWSCC) during the last 25 years. PWSCC of the CRDMs can lead to boric acid corrosion of the reactor vessel closure head (RVCH), resulting in risks of primary water leaks and therefore potentially having a significant impact on the plant safety. As some of the Belgian nuclear power plants are also concerned, it was decided to perform the RVCH and the CRDMs replacement for two Belgian nuclear power units: Doel 4 and Tihange 3. Besides the replacement of these components, the project was also the opportunity to perform the renovation of the rod position indicators (RPIs) and electromagnetic assemblies (EMAs).

In 2011 Tractebel ENGIE (TE) was designated by Engie Electrabel (EBL) as owner’s engineer. The manufacturing and installation of the two new RVCHs and CRDMs was granted to AREVA NP.

Each of the manufactured RVCHs is composed of:

- A forged flange and a hemispherical dome;
- 66 penetrations for the control rods;
- Control Rod Driving Mechanisms;
- New thermal insulation.

After manufacturing of the different parts and a partial assembly in factory, the new RVCHs and CRDMs were transported to site where final assembly tasks were performed.

The replacement of the RVCHs and the CRDMs was performed during the outages of March 2015 and September 2015 for the Tihange 3 and Doel 4 entities, respectively. An ALARA evaluation, consistent with both site specific replacement works, was performed for each nuclear site, together with an intensive follow-up of the activities during the execution phase. This paper gives an overview of the preparatory works for the collective dose estimates and the dosimetric follow-up performed during the outages.

Note that old Tihange 3 and Doel 4 RVCHs were equipped with internals disconnection devices (IDDs) which allow quick refuelling. These parts were cut during previous outages of the units and are out of the scope of this study.

2. *Implementation of ALARA*

2.1 Objectives

The following objectives have been defined in the framework of the replacement project, in correspondence with the general objectives of the outage:

- No work incidents;
- No nuclear incidents;
- No radioactive contamination incidents;
- Minimal collective and individual dose according to the ALARA principle.

These objectives have been communicated to all workers prior to the start of the execution phase of the project.

2.2 ALARA working group

Conform to previous large component replacement projects, e.g. steam generator replacements, an ALARA working group with different people specialised in radiation protection and implementation of the ALARA principle was composed. This group was entrusted with the definition of the nuclear safety and radiation protection actions for these projects at Doel and Tihange sites.

Two important tasks were defined for this working group:

- Before the execution phase, this group defined and checked the possibilities to implement the ALARA principle. They studied, with the cooperation of the contractor, dose rates, organisation of work stations and tasks to be performed to finalize the RVCH replacement. The team also calculated and validated the collective dose estimates for both RVCH replacement projects;
- During the outage, the group ensured the dosimetric follow-up and analysed the radiological conditions (radiation and contamination levels) in the vicinity of the replacement site. They regularly checked the biological protections. Furthermore, a daily verification of the collective and individual doses was carried out to obtain a view on the daily dosimetric evolution of the project and to detect anomalies compared to the estimates. Evolutions and actions were discussed with the owner and the contractor on a daily base. After discussion, the estimates were adapted when important changes in the planning were foreseen and if non-negligible extra contributions would arise due to unforeseen tasks.

3. *Estimation of the collective dose before replacement*

3.1 Measurements and estimates

During the previous outages, dose rate measurements were performed around the RVCH according to Figure 1. This mapping helps to identify hot spots and, in function of the tasks performed in the vicinity of these points, extra shielding could be foreseen to reduce the collective and individual dose uptakes. The measurement campaigns showed that the dose rates for Doel 4 were lower than those for Tihange 3. In average the estimated dose rates were 30% lower for Doel 4.

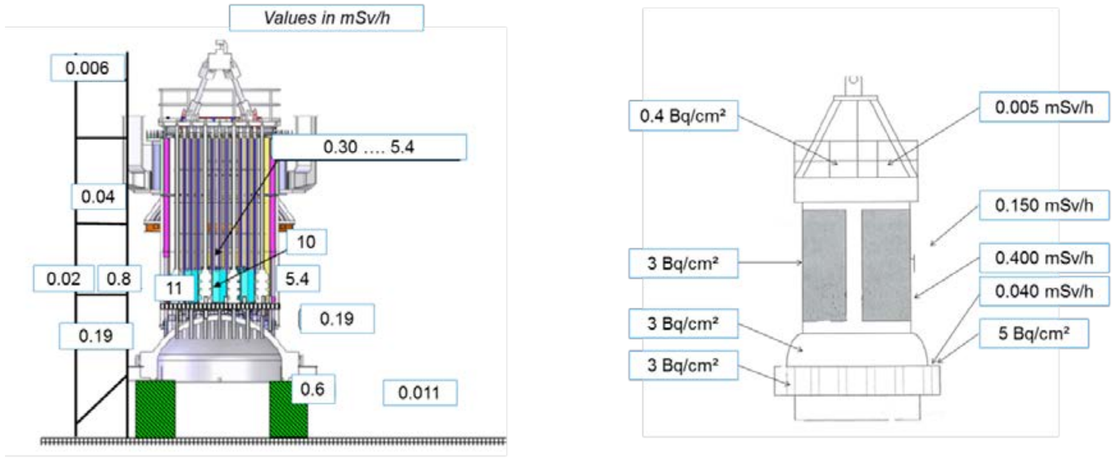

Figure 5 – Dose rate and contamination measurements of the RVCH of Tihange 3 (left) and Doel 4 (right).

These measurement points do not cover all specific work areas. Additionally, dose rate estimates were performed based on the contractor REX in combination with measurements. Due to the similarity between the Doel 4 and Tihange 3 RVCHs, the measurements taken during the execution phase of the Tihange 3 replacement project were used as REX for Doel 4.

3.2 Methodology

Based on estimated man-hours, measured/estimated dose rates and actions taken by the ALARA working group, a collective dose can be assessed. The following methodology is used:

- Definition of dosimetric phases, linked to the project execution phases. A dosimetric phase is a period during which the radiological state (dose rates defined for a specific component) of the working area remains stable (e.g. removal of the ventilation mantle, removal of the EMAs, etc.);
- Subsequently the different tasks and their corresponding work stations are defined;
- Each workstation is linked to its corresponding dose rate estimates (per dosimetric phase). The estimated average dose rate per work station and per dosimetric phase will be collected in a database (DATADOS);
- For each activity, part of the project, an estimation of the man-hours is performed based on the REX of the contractor (and Tihange 3 in the case of Doel 4);
- Every activity is linked to a:
- Category of tasks to determine dose reduction factors (e.g. cutting, scaffolding, etc.);
- Group of tasks to define the ALARA file (e.g. dismantling of the ventilation mantle, dismantling of the EMAs, etc.);
- EBL defined a set of dose reduction factors, based on the operational REX for specific tasks over the years. These coefficients are used to optimize the calculated collective doses and always represent a value lower than unity.

After definition of these parameters, the collective dose for each activity is calculated and a global dosimetric estimate for the replacement project is obtained.

3.3 Results

Figure 2 shows the evolution of the daily estimated collective dose for both Tihange (red) and Doel (blue) projects. Collective doses of 103.7 man.mSv and 73.9 man.mSv were estimated for Tihange 3 and Doel 4, respectively. The decrease in the collective dose estimate for Doel 4 is due to a reduction of 22% of the estimated man-hours, resulting from the REX of Tihange 3 RVCH replacement project and the lower dose rates around the old Doel 4 RVCH compared to Tihange 3 (see § 3.1).

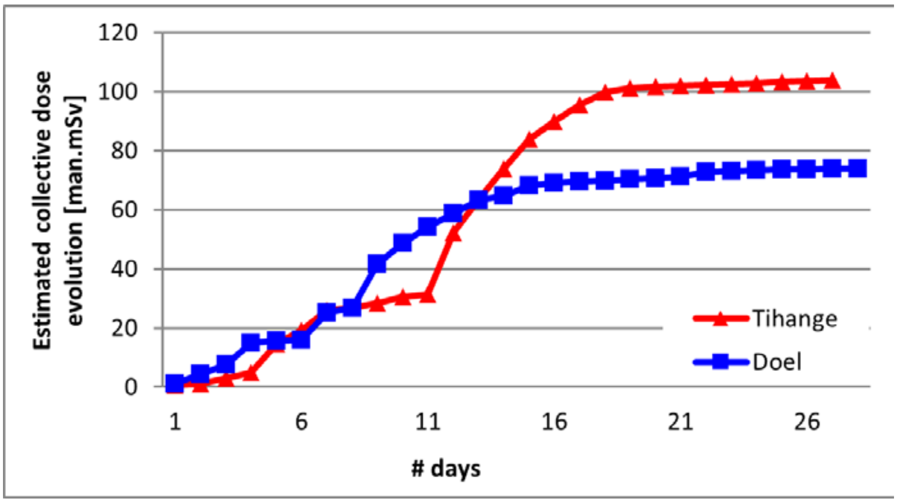

Figure 6 - Initial collective dose estimates for the RVCH replacement projects.

The red curve for Tihange in Figure 2 takes into account the following steps:

- Preparatory works (day 1 to 4): main tasks occur at relatively long distances from the old RVCH which is completely shielded. Consequently, the collective dose is small;
- First dismantling tasks (day 5 to 8): due to the dismantling of the outer ventilation cover and the RPIs, the RVCH self-shielding is reduced and tasks are performed in a higher dose rate area;
- Dismantling of EMAs and introduction of the new RVCH (day 9 to 11) into the reactor building: the removal of the lower shroud and the thermal insulation contributes significantly to the collective dose. Especially the removal of the lower ventilation mantle induces a high collective dose uptake in the proximity of the old RVCH dome and CRDM adapters;

- Dismantling of CRDMs (day 12 to 17) and start of the new RVCH assembly: assembly activities are performed in close proximity of the old RVCH and CRDMs. Consequently, dose rates at the work stations are higher and the collective dose increases significantly;
- End of the new RVCH assembly and evacuation of the old RVCH (day 18 to 27): the old RVCH is extracted from the reactor building and transported in a shielded container. Once evacuated, dose rates around the new RVCH decrease.

The blue curve for Doel in Figure 2 takes into account the following steps:

- Preparatory works (day 1 to 3): preparatory tasks of the Doel project are performed with lower dose rate estimates but are more concentrated. Compared to Tihange, extra tasks were also performed. As a consequence a higher collective dose is achieved in the preparatory phase;
- First dismantling tasks (day 4 to 6): the dismantling of the outer ventilation cover, electrical disconnection and dismantling of the RPIs. As dose rates around the old RVCH are lower compared to Tihange, a lower daily collective dose is estimated. This also explains the less steep slope of the blue curve compared to Tihange;
- Dismantling of EMAs (day 7 to 8): the main EMA dismantling work stations are located at the top level of the RVCH where lower dose rates are present due to the shielding effect of the EMAs and CRDMs;
- Dismantling of CRDMs (day 9 to 14): work stations to dismantle CRDMs are in close proximity of high radiation zones (RVCH dome, CRDM adapters, etc.). Consequently, a fast rise of the collective dose is observed. However this rise is less steep compared to the Tihange project as dose rates were estimated to be lower. Moreover no collective dose is acquired due to the assembly of the new RVCH as it is not introduced in the reactor building at that moment;
- Evacuation of the old RVCH and assembly of the new RVCH (day 14 to 28): the old RVCH is evacuated from the reactor building prior to the new RVCH introduction. Hence the assembly results in a less collective dose uptake.

In general both curves show a comparable evolution of the collective dose. At the beginning of the project a first small increase of the collective dose is observed. This increase begins earlier for the Doel project. Then curves present a second rise of the daily collective dose before becoming less steep until the end of the project. The total duration for both projects is similar. Typically, the Doel project should take more time due to its serial approach (evacuation of the old RVCH before assembly of the new RVCH). However, this increase of time was compensated by a reduction of man-hours.

The second increase is steeper for the Tihange project, especially between day 11 and 17 (day 8 till 12 for Doel). This difference can be partially explained because of the higher dose rates measured at the Tihange power plant. This strong increase can also result from the methodology used for the RVCH replacement. In Tihange the new RVCH was assembled during the disassembly of the old RVCH. Consequently, the dose rates at the work stations of the new RVCH were higher due to the proximity of the old RVCH. For Doel, the new RVCH was only introduced after evacuation of the old RVCH. The collective dose due to the assembly of the new RVCH hence was lower.

4. ALARA follow-up

4.1 On-site follow-up

During the execution phase of the replacement projects, the TE ALARA team acted as a direct link between the TE project team (and its contractors) and the EBL radiation protection team. This allowed efficient and faster communication between the different actors.

The TE ALARA team ensured the following tasks:

- Monitoring and follow-up of the dosimetric estimate in function of important changes in planning or activities that occur during the execution phase;
- Follow-up and feedback on daily and collective dosimetry data;
- Providing feedback and advice for corrective measures;
- Ensuring compliance with rules and instructions that are part of the ALARA procedures;

- Measurements of waste containers leaving the controlled area, prior to transport;
- Measurements of the waste containers after storage.

Figure 3 presents the ALARA procedure which was in application during the execution phase of the project. The radiation protection staff of the owner performed measurements prior to the start of works at the work stations. At the beginning of a dosimetric phase, these measurements were performed together with the TE ALARA team. Figure 3 also lists the different actions that could be considered if unexpected dose rates or contaminations were detected. Inversely, if no contamination or unexpected dose rates are observed, works can start.

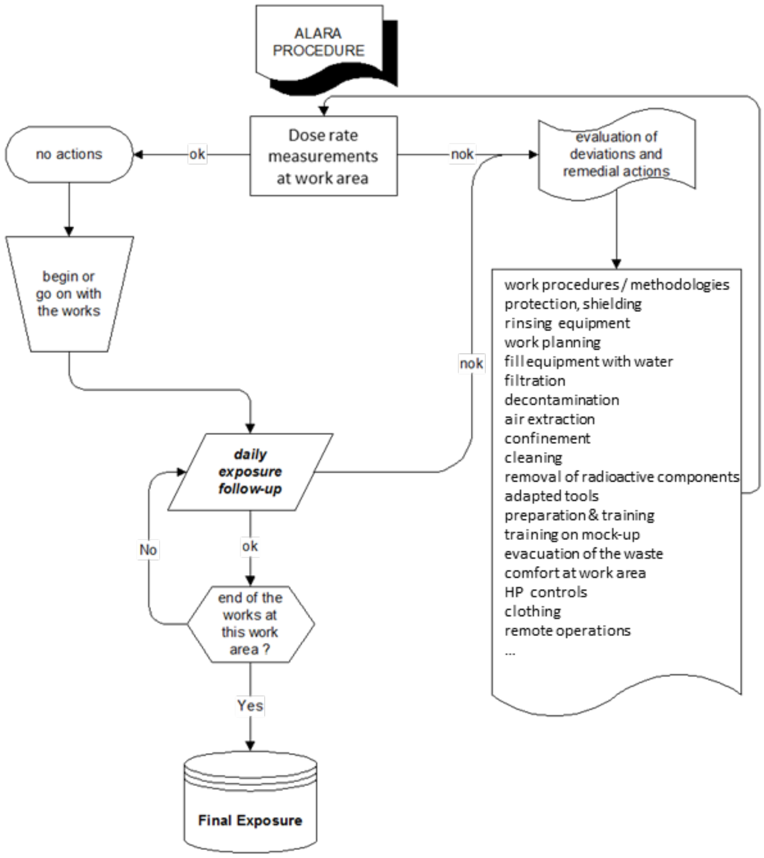

Figure 7 - Flow chart of the ALARA procedure implemented for the replacement projects.

4.2 Recommendations

Both prior to and during the execution phase, the ALARA group gave several recommendations in order to optimize the collective dose:

- The reactor pool was kept filled at all times to minimize the contribution of the reactor core structures to the ambient dose rate levels;
- Cartographies with the dose rates at the level of the scaffolding were presented at the entrance of the working area;
- Additional shielding was foreseen in case hot spots were detected;
- An active sensitization of workers during the execution phase;
- Clear indications at high dose rate areas.

5. Dosimetric follow-up

A daily dosimetric follow-up was performed during the execution phase of the project. It consisted of:

- Daily consultation and processing of individual and collective doses;
- Evaluation of the works progress;

- Regular measurements performed by the TE ALARA team and the radiation protection officers of the owner.

5.1 Tihange replacement project

During the execution phase, some of the EMAs were blocked and had to be manually dismantled in close proximity of the RVCH dome. This resulted in a significant collective dose contribution. Based on estimated man-hours and measured dose rates, a collective dose of 9 man.mSv was estimated for the dismantling of these EMAs. The collective dose after the dismantling of the blocked EMAs was finally limited to 3.62 man.mSv due to optimization by using extra shielding at the scaffolding platform and RVCH dome.

Because of elevated dose rates at the level of the RVCH storage container, a temporary lead wall was built in the storage building to reduce the dose rate levels outside of the storage building. This extra task represents a collective dose of 0.9 man.mSv. Today, the lead wall is replaced by a concrete wall to ensure that site regulations are permanently complied with. The dosimetric results coming from this last action have not been included in the curves but represent a collective dose of 1.2 man.mSv (2 man.mSv was estimated prior the execution phase).

Figure 4 illustrates the collective dose evolution during the RVCH replacement of Tihange 3 (purple curve). The green curve represents an update of the initial estimate prior to the project, taking into account the delays due to the blocked EMAs and other delays that occurred during the project. Compared to these estimates, the purple curve shows daily collective doses lower than the estimated ones. The total project collective dose result is 76.3 man.mSv.

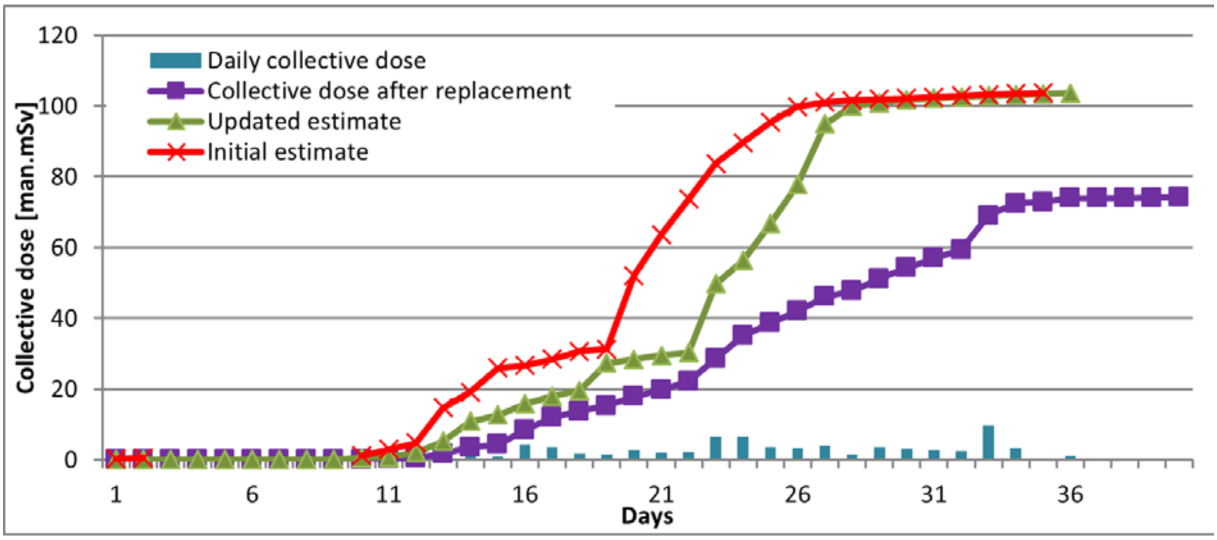

Figure 8 - Comparison between collective dose after replacement and estimated collective dose for the Tihange 3 project.

5.2 Doel

During the execution phase, a delay of one week appeared due to a problem with a bi-metallic weld on the new RVCH. During this period only a limited number of tasks were allowed close to the old RVCH. The only task performed was the removal of blocked EMAs. Based on the limited number of blocked EMAs and the dose rate measurements, no collective dose estimation was performed.

Figure 5 illustrates the collective dose evolution during the RVCH replacement of Doel 4 (purple curve). The green curve represents an update of the initial estimation prior to the project, taking into account the delay due to the issue with the bi-metallic weld. Compared to the estimates, the final collective dose was 52.02 man.mSv/h, i.e. 30% lower than the estimates. Based on the project conditions, this can primarily be linked to the positive influence of lower dose rates at the different work stations.

An important contribution to the collective dose for the Doel project was the decontamination of the RVCH after having found high levels of contamination (wipe samples up to 10 mSv/h in contact). The decontamination, initiated to reduce the risk of further spread of contamination and internal contamination of the workers, represents 6 man.mSv or about 12% of the total collective dose consumed during the execution phase.

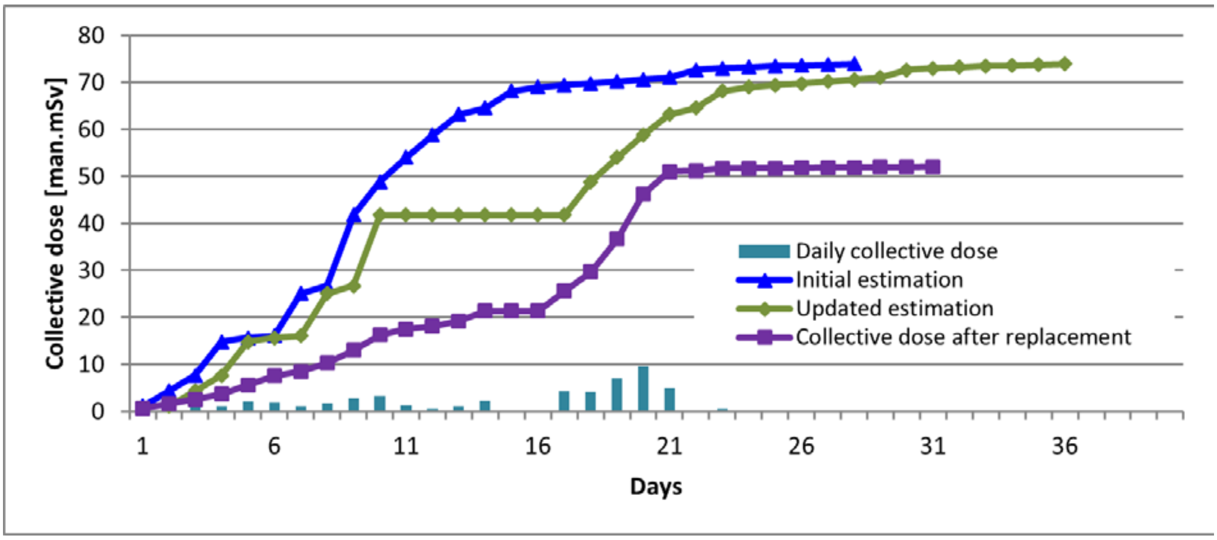

Figure 9 - Comparison between collective dose after replacement and estimated collective doses for the Doel 4 project.

6. Lessons learned

Some conclusions can be highlighted, based on issues that occurred during the execution phase:

- Prior to the transfer of the old RVCH towards its storage container, dose rate measurements around the old RVCH should be performed to determine whether transport conditions will be met. An assessment of the required shielding should be made if dose rates were too high;
- A back-up plan for dose intensive tasks should be available (e.g. for the disassembly of blocked EMAs, issues with the CRDMs cutting device, etc.);
- After cutting of components with high risks of contamination, a radiation protection officer should be informed to check whether no spread of contamination occurred.

7. National and international comparison

7.1 National comparison

In 1999 the RVCH of the Tihange 1 unit was replaced. This project however was significantly different from those performed in 2015, resulting in a significant higher collective dose during the 1999 replacement project.

For the Tihange 1 replacement, the CRDMs had to be re-used. Consequently, a decontamination of the EMAs and CRDMs prior to their installation on the new RVCH was required, resulting in more dose intensive tasks.

The total collective dose for the Tihange 1 replacement project was 275 man.mSv, being a factor 3.6 higher compared to the Tihange 3 project (see also Figure 6).

7.2 International comparison

In Figure 6, a comparison between several international RVCH replacement projects is given (ref. [2], [3] and [4]), starting from the earliest (Bugey 5, France, 1994) to the most recent project (Doel, Belgium, 2015).

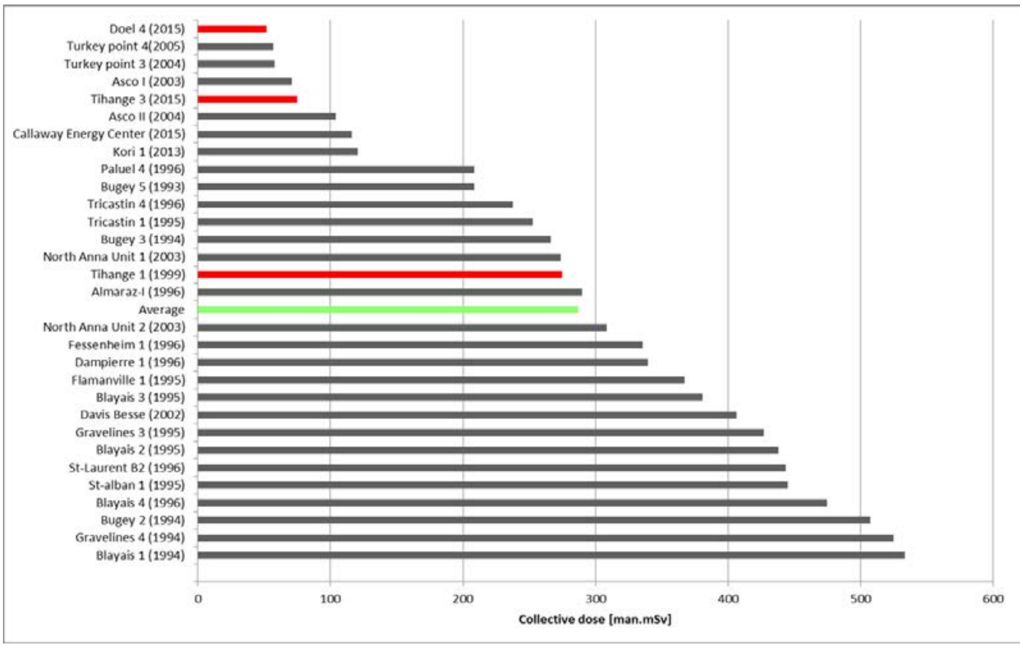

Figure 10 - (Inter)national comparison of RVCH replacement projects. In red, the Belgian replacement projects and in green, the average value of all reported replacement projects [2][3][4].

In general, a decrease in collective dose is observed for more recent replacement projects. This can be explained by:

- REX of previous projects;
- Optimisation of the tasks and techniques, including qualified methodologies that reduce the time necessary for dose intensive tasks (e.g. remotely operated cutting tool for CRDMs).

Comparison between the different projects is difficult because there are significant differences between replacement projects (e.g. serial or parallel approach for the dismantling). Moreover the radiological conditions of the RVCH can vary strongly in function of the concerned nuclear power plant.

8. Conclusion

For both the Tihange and Doel replacement projects, the objectives have been met. The Tihange and Doel projects resulted in collective dose of respectively 76.3 man.mSv and 52.02 man.mSv. The Doel project ended with 32% less collective dose compared to the Tihange project which is explained by the difference in approach, lower dose rates and the REX of the Tihange project.

Based on international data, the dosimetric results of the most recent RVCH replacement projects of Doel and Tihange are in line with expected collective dose results.

9. References

[1] Stress Corrosion Cracking in Light Water Reactors: Good Practices and Lessons Learned, IAEA Nuclear Energy Series, No. NP-T-3.13.

[2] NATC ISOE Information Sheet No. 2003-08 – United States pressurized water reactor (PWR) reactor head replacement dose benchmarking study.

[3] Occupational Exposures at Nuclear Power Plants – Annual reports – ISOE.

[4] ETC ISOE Information Sheet No. 09 – Reactor Vessel Closure Head Replacements (1994 – 1996)

Poster 12. AREVA's Technology for Reduction of Radiation Impact in case of Severe Accident in NPP

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Mitigation of severe accidents in nuclear power plant focuses on protection of the public, the operators and plant structures/systems. Special attention has to be paid to reducing of possible radiation impact. Such technical solutions as filtered containment venting allow significant reduction of impact on environment. Currently available systems cope with wide range of technical requirements.

Using of filtered containment venting has to be supported by monitoring systems which provide detailed information for severe accident management procedures. On-line containment atmosphere monitoring system HERMETIS makes easy an observation of combustible gases inside containment and identify molten core concrete interaction situation. Post-accident sampling system PASS provides detailed information about nuclides available in containment what allows an optimization of severe accident management strategies, e.g. optimal time frame for using of mitigation measures.

An additional task is the protection of plant staff responsible for mitigation of severe accidents. The highest amount of activity impacting the staff in control rooms is represented by active noble gases as Xe and Kr. A solution with unique features (CRAFT) has been developed to reduce the exposure of the plant staff in control rooms. Entering of control rooms by active noble gases is significantly reduced what allows using of the rooms in case of severe accident without time limitation.

Special attention paid that the design of all products allows its easy retrofitting to existing power plants as well as to low installation and maintenance effort.

Poster 13. Implementation of a Radiological Emergency Monitoring System for Bruce Power Nuclear Power Plant (Canada)

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The Bruce Power nuclear power plant (BP NPP) in Ontario, Canada, is the largest nuclear generating station in the world, operating 8 nuclear reactors producing 6300 MW. In correlation with Bruce Power's safety culture, "Safety first" and continuous improvements are essential and substantial parts of the Bruce Power philosophy and management system.

After the Fukushima nuclear accident the Canadian Nuclear Safety Commission (CNSC) released recommendation 5c which asked for the improvement of tools to provide off-site authorities with automated real-time stations, appropriate back-up power, redundant communication channels etc. For BP it became essential to measure not only gamma dose rate, but additionally collect information about the composition of nuclides. Moreover the system should be able to measure and operate reliably even with gamma dose rates up to 1 Sv/h, as seen during the Fukushima event. Besides long-term reliable operation under harsh Canadian weather conditions, a key issue was the safe operation and data transmission during emergency conditions. Consequently the compatibility with seismic requirements, permanent autonomous operation and a reliable redundant communication interface were key specifications requested by Bruce Power.

The tender published in 2012 was divided in two lots: First part comprised a total of 49 fixed and 10 mobile spectrometric stations to which Saphymo responded with the SpectroTRACER probe with LaBr3 (Ce) scintillation detectors with additional GM tube and GPS. This probe is a fully autonomous self-contained gamma spectroscopic measurement station designed to comply with emergency situations also in worst case weather scenario. Battery capacity is available for up to 10 days.

Several communication interfaces are provided: cell/3G modem, Globalstar satellite modem, both integrated and operated over Virtual Private Network (VPN) for secure data transmission. The network interface is always active. Software functions provide an automatic switchover from cellular to satellite. A built-in webserver is available for easy remote control and setup. Besides that several meteorological stations have been integrated. After a several months test run during 2013 all stations have been operational from fall 2014.

In the second part of the project Saphymo delivered 8 SA200NG and SA210ING air monitors. The air monitors have been designed for the detection of ultra-low activities of beta/gamma measurement for particulates, noble gas and iodine with spectroscopic functions and the same redundant data transmission. The SA210ING type air monitors are equiped with a internal power backup for up to 5 days.

Poster 14. EVEREST Project: Entry into the RCA without Specific Protective Clothes

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4 NPPs (nearly 6 at the end of 2016) in France have implemented the EVEREST Project. It aims at enabling the access in the RCA without specific RCA clothes, directly with work overalls. When entering in areas with contamination levels higher than 0.4 Bq.cm-2, workers have to wear specific clothes adapted to the contamination level of the worksite and the type of activity.

This organization allows:
- Reaching and maintaining a high level of radiological cleanliness of the RCA,
- An easier access to the RCA with overalls to perform and/or monitor activities.

After several years of implementation of the EVEREST project on 4 NPPs of the EDF fleet, radiological cleanliness of these NPPs have improved: 70 % of clean rooms in 2006 compared to more than 90 % currently. Workers are more concerned by the contamination in the RCA, when performing their activities and controlling tools and themselves in the field. Workers for routine maintenance activities welcome such working conditions. During outages, due to a large amount of maintenance activities making it difficult to maintain radiological cleanliness at the expected level, workers have to wear specific protective clothes to enter the RB. In contaminated areas of the RB, workers have to wear another layer of protective clothes. In order to improve working conditions in these areas, the plants in collaboration with the EDF fleet engineering division are testing a new approach to only have one protective clothe layer to work in the contaminated areas of the RB.

Poster 15. NATC Utility Data Analysis Working Groups to Achieve New Applications for New ALARA Tools

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NATC ISOE, University of Illinois

The results of a new initiative recommended by US RPMs to conduct monthly NATC CZT data analysis and isotopic mapping results will be presented. Over 35 utilities participated in the working group to better understand the best uses of new CZT technology and share the results with ISOE members. Over 25 new and remarkable new "discoveries" regarding nuclear plant radiological environment are described based on case studies submitted by Prairie Island, Palisades, Cook, Bruce and Palo Verde.

Poster 16. Radiation Protection Features related to the Processing and Packaging of Plutonium in GTRI Project

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The mission of the Global Threat Reduction Initiative (GTRI) is to reduce and protect vulnerable nuclear and radiological material located at civilian sites worldwide. GTRI supports the U.S. Department of Energy’s nuclear security goal by preventing terrorists from acquiring nuclear and radiological materials that could be used in weapons of mass destruction or other acts of terrorism.

Inside the GTRI project and via a joint effort between the Department of Energy (DOE) and Italy’s Società Gestione Impianti Nucleari (SOGIN), the thirteenth shipment from Italy to the United States of material such as highly enriched uranium (HEU) and plutonium was completed. These materials were located at three SOGIN facilities in Italy.

The aim of the paper is to assess the radiation protection issues and to analyze the measures taken to minimize the radiological risks for the personnel involved in the following activities related to the processing and packaging of plutonium, performed in the Plutonium Plant - Casaccia Research Centre (CR Casaccia):

- Nuclear material displacement from the Nuclear Storage to the processing room;
- Matrix preparation, milling and heat treatment for stabilization;
- Thermogravimetric analysis (inside Glove Boxes);
- Radiometric Characterization;
- Repackaging of Nuclear Materials;
- Nuclear material displacement from the processing room to the Nuclear Storage;
- Loading the container for shipping.

Before starting to work, it was assessed the radiological risk for the workers in normal, abnormal and accidental conditions.

To carry out the activities were engaged a total of 38 workers (n. 5 Operators, n. 12 Operator Assistants, n. 12 Radiation Protection Technicians, n. 4 Supervisor and n. 5 Radiological Characterization Technicians) to complete the task in 70 days (daily mean values of workers involved in the task: n. 3 Operators, n. 4 Operator Assistants, n. 4 Radiation Protection Technicians, n. 2 Supervisor and n. 1 Radiological Characterization Technicians).

To test the effectiveness of the devices and of radiation protection techniques, were performed the following investigations for a total of 2800 measures:

- Detecting removable surface contamination of the components/equipments and of the workplaces by means of alpha/beta counting of smears;
- Measurements of activity in nasal smears for workers involved in risk activities;
- Readings of direct reading dosimeters for workers involved in the task;
- Alpha/beta counting of alpha continuous local air monitors’ filters (ALPHA-7 with warning and alert thresholds);
- Alpha/beta counting of air sniffers’ filters installed around Glove Boxes;
- High-resolution gamma-ray spectrometric measurements of air sniffers’ filters;

- Alpha/beta counting of continuous air monitors' filters related to workroom;
- Alpha/beta counting of continuous air monitors' filters related to Glove Boxes and to inner and outer chimneys;
- "In vivo" measurements of body (Whole Body Counting – for high energy photons) and of lung (Lung Counting – for low energy photons) content of radionuclides;
- Excreta monitoring (alpha spectrometric measurements of urine and faeces to determine Pu-238, Pu-239+240 and Am-241 contents).

Analyzing the results of these investigations can summarize:

- The workplace monitoring (sampling air and surfaces) gave values lower than the minimum detectable activity (MDA);
- The individual monitoring for internal exposure of workers (direct and indirect measurements) put in evidence some positive cases (below 1.3 mSv of internal dose, though not directly related to the GTRI project);
- The highest daily radiation dose derived from the readings of direct reading dosimeters was 29 µSv; the highest individual cumulative dose for all the work period was 0.24 mSv, far below the estimated value (about 2.5 mSv).

On the basis of the above informations, the radiological risk for the workers involved in the task hasn't been significant if compared to their classification. The procedures adopted and the types of personal protective equipment (PPE) used showed the effectiveness and appropriateness of the devices and of radiation protection techniques.

Poster 17. VAHA, a New Requirement Managements System used in the NPP Sector in Finland

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According to Section 7 r of the Nuclear Energy Act (990/1987), the Radiation and Nuclear Safety Authority (STUK) shall specify detailed safety requirements for the implementation of the safety level in accordance with the Nuclear Energy Act. The requirements issued by STUK are presented in the YVL Guides.

When considering how the new safety requirements presented in the YVL Guides shall be applied to the operating nuclear facilities, or to those under construction, STUK will take due account of the principles laid down in Section 7 a of the Nuclear Energy Act (990/1987): The safety of nuclear energy use shall be maintained at as high a level as practically possible. For the further development of safety, measures shall be implemented that can be considered justified considering operating experience, safety research and advances in science and technology.

According to the Finnish Nuclear Energy Act, the safety requirements of STUK are binding on the licensee, while preserving the licensee's right to propose an alternative procedure or solution to that provided for in the regulations. STUK may approve the alternative procedure or solution by which the safety level set forth is achieved.

Today there are 45 regulatory guides issued by STUK in the nuclear sector. Seven of these YVL Guides consider Radiation Protection. In the YVL Guides there are altogether more than 6000 different requirements.

In order to better manage and follow the fulfillment of the different requirements of the YVL Guides a new computer based management system called VAHA was created. The purpose of VAHA is to attach specific attributes to different requirements in order be able to better focus a certain requirement to the different life-phase of a nuclear facility.

The new requirement management system is of special necessity in Finland because there are different phases of lifecycle of nuclear facilities present at the same time. There are four NPP units in operation, one NPP unit is nearing its operation license phase, one is in the construction license phase and one nuclear research facility is about to reach its decommissioning phase.

The presentation will describe the use of VAHA.

Poster 18. Localisation of the Source of Airborne Contamination in EDF Reactor Building in Real Time by using CFD Simulations

A., Ferrer, G., Le Meur, E. Gaillard-Lecanu & S. Jahan

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Airborne contamination control is a key issue during outages to protect efficiently workers. On EDF NPPs, internal contamination is not tolerated. Consequently, EDF has long striven to improve the monitoring device and knowledge of the phenomena to be taken into account.

In that context, EDF R&D worked in partnership with IRSN, in the beginning of the 2000's to develop a model of aerosols transport and deposition to simulate contamination dispersion and help to optimize contamination detection monitoring (CAM's) in reactor building. This model is based on an Eulerian approach, coupling a transport equation and an original wall law and enables to optimize the number and location of CAM's during outages. This model is implemented in the EDF open source CFD calculation code Code_Saturne.

EDF R&D has kept on working on this subject to localize in real time the contamination source by using this CAM's outputs. There are some methods in literature to realize this inverse calculation but very few in real-time. Indeed, a full RB calculation needs more five days to run. The methodology presented here is based on the correlation between the real-time contamination CAM's measurements and references scenarios previously simulated with Code_Saturne. This approach was validated by aerosols and gas tracer experiments in the reactor building of Golfech NPP. In the future, a tool based on this methodology could greatly help the radiation-protection department to localize contamination sources in case of incidents.

Poster 19. Radiation Protection for Tritium from Qinshan NPP III in China

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There are two heavy water reactors (CANDU 720MWe×2) in commercial operation in Qinshan NPP III. The tritium releases from heavy water reactors are significantly higher than those from pressurized water reactors. A large amount of tritium is produced in the D2O coolant and moderator, primarily in the operation of the moderator system. Tritium internal exposure can contribute about 10-30% of total collective dose in the occupational exposure. Thus, the control and protection for tritium is very important for CANDU reactors.

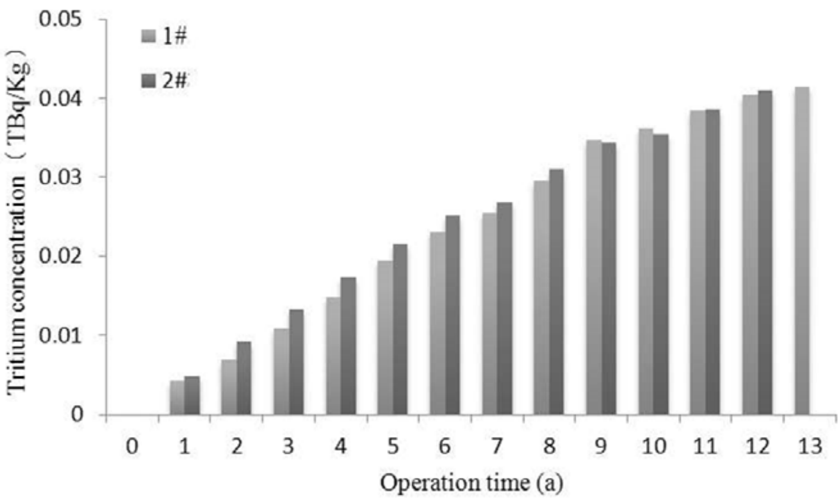
1. Tritium concentration control in workplace

（1）Reduction of D2O leakage

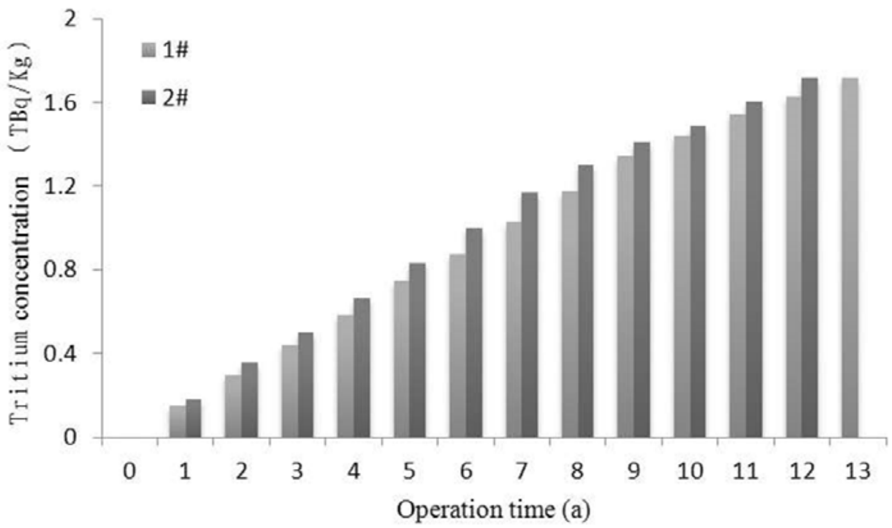
The thermal neutron flux in the CANDU reactor is the major producer of tritium by the activation reaction, 2H（n,γ）3H. The reaction occurs in the D2O of both the moderator and heat transport systems.
CANDU reactors in Qinshan NPP III have operated for more than 10 years. Tritium activity concentrations in the D2O of both the moderator and heat transport systems are up to about 1700 GBq/L and 40 GBq/L, respectively. Generally, the heat transport system has a much higher D2O leakage rate than the moderator system and its D2O leakage can contribute about 50% to the tritium dose to operators. Due to high tritium concentration, even small D2O leakage in the moderator system can result in unexpected tritium intake to operators.

（2）Tritium removal of heavy water

Tritium concentration in D2O of both the moderator and heat transport systems gradually increases as a function of the reactor operating time.



Tritium concentration in D_2O of heat transport system



Tritium concentration in D_2O of moderator system

After some years of operation, there is much high tritium concentration in the D_2O. It is necessary to make appropriate measures to remove tritium of D_2O, especially in moderator transport system. This is beneficial to decrease tritium concentration in the workplace.

（3）Recovery of heavy water

The D_2O leakage from the heat transport and moderator systems is collected and either it is stored and sent to the respective systems or upgraded first if needed before the transfer. The reactor building is divided into four separate areas containing atmospheres of different D_2O and tritium isotopic contents. Each area has been provided with separate D_2O recovery and collection equipment to prevent unnecessary downgrading or tritium contamination. Thus, although the D_2O leakage is inevitable, the heavy water management systems serve as a means of control within the plant and as one or more of the barriers to their release to the environment.

2. Individual monitoring for tritium internal exposure

（1）Urine tritium monitoring

In Qinshan NPP III, urine tritium monitoring to workers includes routine monitoring and special monitoring. For routine monitoring, monitoring period is fortnight or month.

Monitoring periods	Month	Fortnight	Week	Day
Investigation level	52 Bq/ml	184 Bq/ml	260 Bq/ml	400 Bq/ml
Intervention level	6 mSv			

Radiation protection staff must check the reason when urine tritium concentration is above investigation level. The intervention measures must be performed when the tritium dose is expected to above 6 mSv.

（2）Tritium internal exposure

For CANDU reactor, tritium is important resource of dose to operators. In Qinshan NPP III, tritium can contribute about 10-30% to total collective dose to operators. Record level for tritium dose is set to 50 μSv.

Collective dose to operators in Qinshan NPP III

Operation time	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014
Total dose (man·Sv)	0.171	0.81	1.368	0.519	0.572	0.788	0.748	0.727	0.832	0.689	0.630	0.708
Tritium dose (man•Sv)	0.018	0.062	0.118	0.146	0.099	0.199	0.163	0.197	0.188	0.188	0.141	0.140
Tritium ratio (%)	10.5	7.7	8.6	28.1	17.3	25.3	21.8	27.1	22.6	27.3	22.4	19.8

Poster 20. AISense Gamma - the First Handheld Instrument with Continuous Real-Time Display of Gamma-Ray Field Direction

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Properties of the first compact instrument with continuous display of both gamma-ray field intensity and its incoming direction across the full solid angle are presented. Intuitive usage and effectiveness of the device change the approach to hotspot localization and vastly improve workflow.

Principles and use cases, including its use at the Krsko Nuclear Power Plant in Slovenia, are discussed.

Poster 21. Analysis of Personnel Radiation Monitoring Results during NPP Outages

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The greater part of collective doses is received by NPP personnel during scheduled and unscheduled outages. Thus, optimizing radiation protection to reduce radiation dose rates should include above all organizational optimization of works performed during outages (planning, preparation, performing and monitoring). For this purpose specific dose limit quotas (budget) have been established for personnel involved in repairs prior to the start of scheduled outage at each NPP operated by the Company. The collective dose quotas have been established for each unit, as well as for main subdivisions invloved in repairs. The preliminary forecasts of collective dose rates have been developed based on radiation survey of nuclear facility equipment during previous outages with regard to future labor input in repairs.

Radiation measurements and assessment of radiation situation have been performed in accordance with the established techniques at all stages of preparation and during repairs of radiation hazardous equipment. Map tables habe been filled in based on results of radiation survey readings of radiation hazardous equipment of the primary circuit.

The planned and actual dose rates for 15 operating NPP units during 2014 outages are provided in the Table below.

Table. Planned (quotas) and actual collective dose rates of NPP personnel during 2014 outages by NPP unit (actual data is provided based on field dose meters), man·mSv

NPP		Unit №1	Unit №2	Unit №3	Unit №4	Unit №5	Unit №6	Total per NPP
ZNPP	*Quota*	*686*	*564*	*530*	*527*	*651*	*542*	*2847*
	Actual	414	519	305	293	-	302	1834
RNPP	*Quota*	*725*	*818*	*767*	*301*	-	-	*1886*
	Actual	-	809	429	199			1437
SUNPP	*Quota*	*800*	*1100*	*850*	-	-	-	*2750*
	Actual	467	644	254				1365
KhNPP	*Quota*	*1100*	*300*	-	-	-	-	*1400*
	Actual	683	486					1169

The collective dose rates and duration of repairs per one NPP unit of NNEGC Energoatom for 2010 - 2014 are provided in the Figure below.

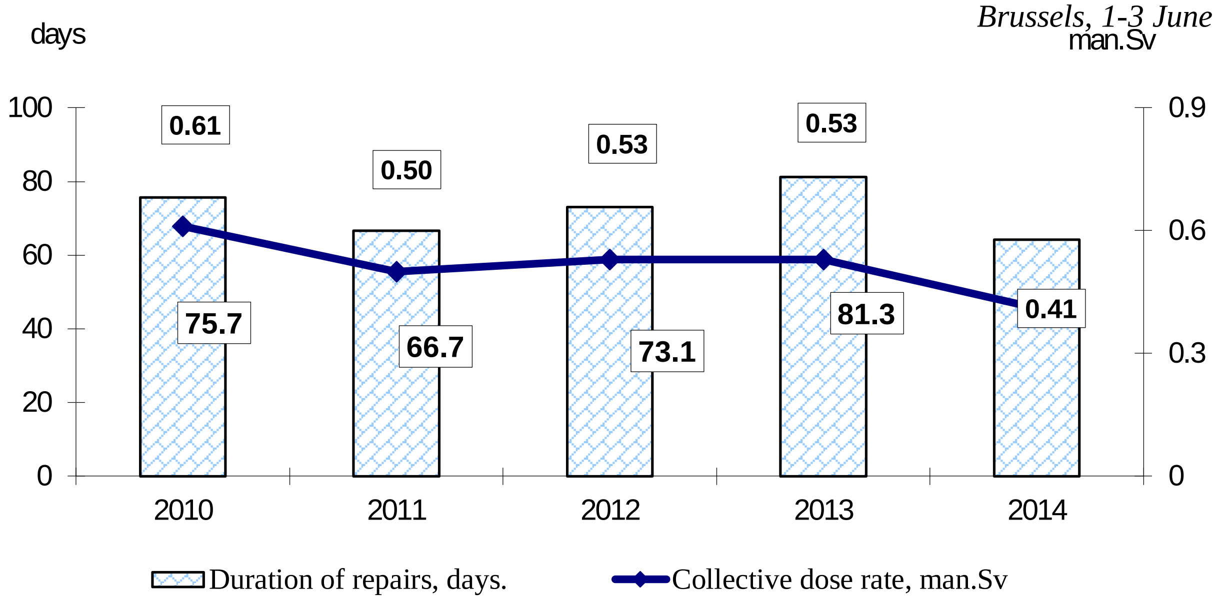

Fig. – *Personnel collective dose rates and duration of repairs per one NPP unit of NNEGC Energoatom for 2010 - 2014.*

Total collective dose rate of NPP personnel involved in repairs of NPP units during 2014 outages within the Company made up 5800 man·mSv, with 4318 man·mSv – for main systems.

The average contribution of separate groups of major equipment to the total collective dose received by NPP personnel during 2014 outages within the Company made up: reactor facility – 19.4%, main circulation pipeline – 5.9 %, reactor coolant pump – 4.2%, steam generator – 17.2%, pressurizer – 2.1%, cooling systems – 9.7 %, auxiliary systems of reactor building – 6.2 %, primary purification system 1.2 – 3.2 %, nuclear fuel (refueling) – 0.8 %.

Therefore, just as in previous years, in 2014 the preliminary forecast of collective dose rates has proved to be correct in practice. More than 50% of the total annual collective dose rate has been received by NPP personnel of NNEGC Energoatom during repair of reactor facilities and steam generators.

Poster 22. Analysis of the dynamics of the collective dose values of the personal of Armenian NPP for the entire period of operation

K. PYUSKYULYAN, M. KIRAKOSYAN V. ATOYAN

Armenian NPP

The paper presents data on the collective dose of the personnel for the entire period of operation (1977 - 2015). The analysis of the ratios of the value of the collective dose of personnel during the period of planned outage and the annual collective dose was implemented.

A significant decreasing of the annual collective dose for the last 15 years is noted. The average value of annual collective dose for the period from 1977 to 1988 is 5.507 man.Sv, for the period from 1998 to 2015 – 0.87 man.Sv. That is, the annual collective dose decreased in 6.33 times.

Between 1989 and 1994 the station was in the extended shutdown mode. From 1994 to 1997, at the station were carried out intensive work to resume operation of Unit 2, so this period is not typical for the normal operation mode.

The reasons for such reduction were considered (application of the ALARA principle):

- Analysis and assessment of estimated work,
- Taking into account of experience gained,
- The doses planning and involvement of personnel in this process,
- Use of additional protective measures,
- Improving the quality of monitoring of radiation exposure and, due to this, implementation of more stringent admission of personnel to controlled area.

Poster 23. Lessons Learned from the Failure of Leak Monitoring System of Steam Generator

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Steam Generator Leakage Monitors are used to detect leakage from primary to secondary cooling system by measuring 6.13MeV gamma rays emitted from N-16.

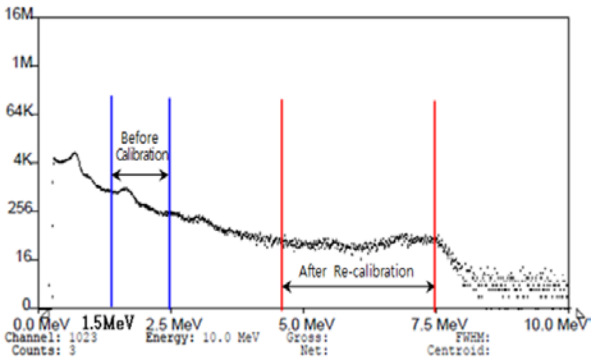
A Cm-244/C-13 sealed radioactive source, which emits gamma rays of 6.13MeV, is used to calibrate the N-16 Channels. The Steam Generator Leakage Monitors use a scale factor to convert count rates of N-16 channel to leak late.

Each detector of Steam Generator Leakage Monitors faces main steam line and mounted in a shielding assembly to protect the detector from the desensitizing of ambient background radiation.

Securing the reliability of the N-16 measurement is of importance to ensure the safe operation of nuclear power plants and minimize the effluence of radio-active isotopes to the environment.

In October, 2014, approximately a 20 l/hr of leakage of the primary to secondary cooling system took place at Hanbit Unit 3. However, all of N-16 monitors didn't work properly during the leakage while other monitors except those worked well.

Therefore, its root cause was analysed and proper actions were taken to prevent the recurrence of the same events. The main cause of the event was found to be the improper calibration procedure regarding N-16 monitors and the failure of a temperature sensor of a detector. The normal N-16 channel range is from 4.5MeV to 7.5MeV, but the spectrum was shifted so that it could not detect 6.13 MeV gamma rays.



[Figure] N-16 channel spectrum

Korea Hydro and Nuclear Power Central Research Institute (abbreviated KHNP-CRI) analysed the cause of the spectrum shift and revised the calibration procedure. It was confirmed that all Steam Generator monitors in domestic nuclear power plants work properly by recalibration in accordance with the revised calibration procedure and their reliability was shown by the inter-comparison tests

Poster 24. Evaluation of Tritium Release from Framatome NPP Using NUREG-0017

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Background

Annual gaseous tritium release from Hanul unit 1&2, Framatome type NPP in Korea, is approaching to limit in FSAR. Therefore, existing evaluation method was reviewed and tritium release was re-evaluated using NUREG-0017, PWR-GALE Code [1].

Method

In FSAR of Hanul unit 1&2, normal and abnormal operating conditions were assumed to evaluate gaseous and liquid tritium release. In case of the normal operating condition, the assumptions were based on operating feedback from French Power Plants and the evaluated values are known as “expected values”. In case of the abnormal operating condition, the evaluated values are known as “design values”. Expected value and design value of liquid tritium release for one unit are 600 and 750 Ci/year, whereas expected value and design value of gaseous tritium release for one unit are 60 and 90 Ci/year, respectively [2]. In case of Hanul unit 1&2, the annual release limits for liquid and gaseous are 750 and 90 Ci/year, respectively.

The PWR-GALE Code is a computerized mathematical model for calculating the releases of radioactive material in gaseous and liquid effluents from PWRs and used when licensing new NPP in USA. The calculations are based on data generated from operating reactors, field and laboratory tests, and plant-specific design considerations incorporated to reduce the quantity of radioactive materials that may be released to the environment during normal operation, including anticipated operational occurrences [1].

Current operational data such as thermal power level, mass of coolant and etc. was used to calculate the releases of radioactive material in gaseous and liquid effluents. The pathway for shim bleed, equipment drain waste, clean waste and dirty waste are 1.0, 0.1, 1.0 and 1.0, respectively. Re-evaluated annual gaseous and liquid tritium release are 110 and 990 Ci/year, respectively. Annual gaseous tritium release is evaluated to 22% greater than the release from existing method whereas annual liquid tritium release is evaluated to 32% greater. The tritium release from existing method was somewhat underestimated than PWR-GALE Code.

Future works

Using re-evaluated data, the radioactive concentration of the effluents at the boundary will be estimated to verify the concentrations are within Nuclear Safety and Security Commission Notice limits. Also, the dose to the general public will be estimated using K-Dose 60 Code, dose assessment program for public around NPP. After estimating concentration and public dose, the licensing document will be presented to Korean Institute of Nuclear Safety to revise the annual tritium release limits of Hanul unit 1&2.

References

[1] US Nuclear Regulatory Commission, Calculation of Release of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-Gale Code), NUREG-0017, Rev.1, 1985

[2] Final Safety Analysis Report, Hanul Unit 1&2, Korea Hydro and Nuclear Power Co., Ltd.

Poster 25. Strengthening Occupational Radiation Protection in nuclear facilities through the IAEA projects funded by the European Commission

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The protection of workers in nuclear facilities is an important part of the IAEA programme on occupational radiation protection. To develop the standards for the protection of workers from occupational exposures and to provide assistance for the application of the standards in the Member States are the main tasks of the programme. As part of strengthening the occupational radiation protection in nuclear facilities under the support of the European Commission, two projects were implemented regarding radiation protection aspects of the decommissioning of nuclear facilities: workplace monitoring and radiation protection of itinerant workers.

The decommissioning document provides practical guidance for the management, planning, and conduct of occupational radiation protection in decommissioning. The need for safety assessments prior to undertaking the decommissioning activities and graded approach in safety assessment is emphasized. Radiological hazards, non-radiological hazards, training and sharing of technical information are emphasized. Optimization of radiation protection, which is the driving force for radiation protection principle, is required to be implemented in all stages of nuclear facilities, including decommissioning. Engineering control measures such as containment of radioactive dust generation, appropriate air changes in workplace for ensuring good hygiene and shielding of external radiation are recommended in order to keep the collective dose and individual dose under control.

The training package on workplace monitoring in nuclear power plants and other radiological facilities has been developed and completed by the IAEA. Recent detection technology such as silicon diode based detector and CZT detector based gamma imaging for contamination monitoring are included. An adequate focus has been provided in external dose rate monitoring describing the objectives, methods, operability and calibration of radiation instruments covering gamma, neutron and beta dose rate monitoring as well. Surface contamination measurements and the real time monitoring air activity measurements in workplace are illustrated. A TECDOC describing the practical guidance on the implementation of effective workplace monitoring in nuclear facilities has been prepared. This document addresses the requirements of workplace monitoring and monitoring techniques adopted therein. It dwells upon the operational quantities, designing a monitoring programme including practical considerations. It emphasizes the need for identifying the location and frequency of monitoring, methods and investigation levels. Specific radionuclide monitoring such as iodine, noble gases, tritium and plutonium are outlined. This supports the occupational radiation protection safety guide.

Itinerant workers are not the employees of a particular organization but may be exposed to sources of ionizing radiation at different facilities on migration. A safety report prepared by the IAEA, addresses the radiation protection issues associated with the employment of itinerant workers, and recommends the application of managerial and practical arrangements that need to be in place if good practices are to be followed and radiation doses to be controlled adequately. To ensure the competence of itinerant workers, there is a need for adequate qualification and training and these elements are emphasized. Further, radiation protection issues which are unique for NPPs and other nuclear facilities are also being addressed in the safety report.