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BOOK OF ABSTRACTS

ISOE European Symposium

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DOSE REDUCTION PLANNING AT EXELON NUCLEAR

Willie Harris, CHP
Director, Radiation Protection
Exelon Nuclear
willie.harris@exeloncorp.com

In 2010, the Corporate Radiation protection undertook a project to reduce the overall fleet exposure at the 17 Nuclear Units owned and operated by Exelon Nuclear. At the time, there was one unit in the "Top Quartile" rankings published by the Institute of Nuclear Power Operations (INPO) and four units achieving full INPO points. Corporate radiation protection developed procedures, processes, and controls to improve the overall collective radiation exposure at the sites and at a fleet level. As a result, at the end of 2013, the fleet has shown significant improvement in collective radiation exposure with 13 of the 17 units obtaining full INPO points and 9 of the units in top quartile performance. This presentation will discuss the procedures and processes established to reduce collective radiation exposure from a fleet and individual site perspective.

ORGANISATION OF AN ALARA BENCHMARKING IN SWEDISH NUCLEAR POWER PLANTS

Karin Fritioff [a], Marie Carlson [b], Torgny Svedberg [b], Staffan Hennigor [c],
Sylvain Andresz [d], Caroline Schieber [d]

[a] Vattenfall Research and Development AB, Stockholm, Sweden
[b] Ringhals Nuclear Power Plant, Sweden
[c] Forsmark Nuclear Power Plant, Sweden
[d] Centre d'étude sur l'Évaluation de la Protection dans le domaine Nucléaire (CEPN), Fontenay-aux-Roses, France

The optimization principle is a comprehensive and iterative approach for using protection and safety measures to make radiation exposure, and the probability and magnitude of potential exposure, As Low As Reasonably Achievable or ALARA. Ringhals NPP and Forsmark NPP have always worked according to the optimization principle and have developed comprehensive work management programmes and processes to achieve its objectives. For example, the commitment of managers and staff in the radiation protection field was emphasized by the IAEA Operational Safety Review Team (OSART) Report after a mission in Forsmark NPP in 2008.

In 2011, the Swedish Authority (SSM) performed specific ALARA inspections in Forsmark and Ringhals NPPs to review the general organization of the optimization of worker's protection as well as the integration of the optimization principles in plant's modification projects and jobs. These inspections offered a number of proposals for improvements and progress.

In response to these inspections RadNet, the Vattenfall's Radiation Protection Network, considers the possibility of an ALARA Benchmarking visit on its Swedish NPPs. Basically, the aim of this benchmarking is to help answering these questions: *With the management used for the process of optimisation of the worker protection at our plants, with the objectives in the occupational radiation protection area, are we keeping the doses as low as reasonably achievable? Or could another set of activities, another set of targets/objectives, another focus of ALARA-program, be more efficient and favourable for the occupational doses and radiation fields at our plants?*

CEPN, with thirty years of international experience in the elaboration of methodological concepts related to ALARA and its practical implementation, has been chosen to perform the ALARA Benchmarking. This benchmarking visit will be performed in March 2014. It will involve not only CEPN team, but also two Radiation Protection Managers from other NPPs well known for their good radiation protection results and successful ALARA Programmes.

The presentation will first describe the ALARA Benchmarking Guide, elaborated by CEPN, in cooperation with Vattenfall. It will then focus on the ALARA Benchmarking results and lessons learned.

RINGHALS – ALARA PERFORMANCE INDICATORS

Madelene Johansson
Ringhals AB
43285 Väröbacka
Sweden
+46 340 785830 (office phone)
+46 730 581753 (work cell phone)
madelene.johansson@vattenfall.com

The nuclear industry is continuously being challenged by the ALARA concept and ways to minimize dose. To set up goals and find best practices plants are encouraged to compete with each other and measure this performance by plant collective radiation exposure (CRE) comparisons. However there are many other factors that need to be taken into consideration when comparing CRE between plants, such as plant design, source term, operating cycles, NSS vendor, shut downs etc.

The question is: How do we measure ALARA performance and CRE? Are there alternative ways to break out the CRE so it makes sense from a plant specific perspective for comparison?

This presentation will give some examples using Ringhals data compared to the rest of the fleet using available data in the ISOE and WANO database.

EDF "GRAND CARÉNAGE" PROGRAM: IMPACT ON THE COLLECTIVE DOSE AND SUBSEQUENT ALARA ACTIONS?

François DROUET[1], Christine PIGNOT[1], Serge BLOND[2]
[1] EDF/DPN/UNIE/GPRE – 1 place Pleyel – 93282 Saint-Denis Cedex – France
+33 1 43 69 06 18 – francois.drouet@edf.fr
[2] EDF/DPN/État-Major Radioprotection – 1 place Pleyel – 93282 Saint-Denis Cedex – France

In 2011, EDF launched the preparation of the "Grand Carénage" program, which main goals are to implement the technical conditions to operate the French reactors over 40 years and to integrate the consequences of the Fukushima accident. This program is planned to be implemented between 2015 and 2025.

The "Grand Carénage" program implies a significant increase of maintenance and modification activities in particular those performed in the radiation controlled area. Examples of significant dosing activities are the following: steam generator replacements in the 1300MWe reactors, pressurizer heater replacements, replacement of the CVCS heat exchangers, multiplication of valves activities, etc. Subsequently the increase of radiation controlled area activities also implies the increase of logistic and radiation protection works. As a consequence, a significant increase of the collective dose per reactor is expected when the "Grand Carénage" program will be fully implemented.

The objective of this presentation is to expose the evaluation of the impact of the "Grand Carénage" program on the collective dose per reactor on the French fleet and the actions launched by EDF in order to limit the increase of the collective dose.

IMPROVEMENTS IN ALARA MANAGEMENT AT CERNAVODA NPP

Vasile Simionov, Vasile.Simionov@cne.ro **; Ion Popescu,** Ion.Popescu@cne.ro

"CNE Cernavoda" NPP, No. 2, Medgidiei Str. Cernavoda 905200, Romania

Cernavoda NPP management is committed to promote an ALARA policy in order to maintain the organization in the top RP performances. The present paper presents the main results of Cernavoda NPP concerning the three key focus areas of Big RP, an industry initiative to drive excellence in radiological safety:

1. ***Effective management of the source term (sources of worker dose)***: tritium is the main concern of Radiation Protection in a CANDU plant, being responsible for up to about 30% of the collective exposure. CNE Cernavoda developed a station policy to improve tritium dose control, which contributed to a major reduction of internal exposure. Since the heavy water is also an important resource, economically thinking, its careful management is important. Tritium in air activity surveillance using Tritium in Air Monitoring System (TAM) it proved to be an efficient tool for early identification of heavy water spills and internal exposure optimization. When tritium concentration exceeds TAM alarm thresholds the system generates alarms for high level of tritium fields: On February 2012, TAM system identified high tritium level in a reactor building room. A heavy water leakage was discovered on main PHT pump. This leakage had been monitoring through TAM system in order to quantify the leakage level until Planned Unit 1 Outage. TAM System air sampling and measuring during the expected periods of increased concentration, allowed to determine the need for engineering controls, respirators, area evacuation, area posting, and worker relief from unnecessary respirator use. The evolution of radiation fields during the outages is studied since 2010 in collaboration with Kinectrics. The up to date results indicates that the deposits of activation and fission products are significantly lower than other CANDU plants. This is due to a good chemistry control and a proper operation of primary heat transport and moderator circuits. Hot spots management program (identification, characterization, removal or shielding) is in place.
2. ***Increasing organizational and individual ownership and accountability for work processes to manage radiological risk effectively***: administrative controls, responsibilities, and duties for direction, control, conduct, and oversight of radiological risk significant work inside Radiologically Controlled Area (RCA) are well established.
3. ***Reducing collective and individual radiation exposure***: Five years dose reduction plan was developed and approved by senior management to provide oversight and resources for dose reduction initiatives. Since 2009 to 2012 the collective dose per operating unit al CNE Cernavoda had values between 196 and 459 man·mSv performance which classified our plant in the top of CANDU plants. A comprehensive analysis was done in 2012 to identify collective dose reduction possibilities and an action plan was issued.

NEW ELECTRONIC DOSIMETER SYSTEM AND ELECTRONIC PERSONAL DOSIMETER AT OSKARSHAMN NPP, SWEDEN

Mats Hjelm and Catrin Bauréus Koch

Dept. Health Physics, Oskarshamn NPP, Sweden

Introduction: A new work dosimeter system was implemented at OKG NPP in 2010. At the same time a new electronic personal dosimeter was taken into use.

Material and method: EDOS (Electronic DOse information System) is a system developed from the last thirty years of experience from the former ARBDOS (Work Dose information System) system. The EPD (Electronic Personal Dosimeter) is especially designed to meet the complex requirements regarding isotropy, broad energy interval, detection level and ability to secure the use of dosimeter inside the controlled area.

Discussion: The monitoring of the workers has considerably increased since it is now possible to reconstruct the dose and dose rate in intervals of 1 s, thereby knowing the exact dose given to the worker at a certain object or if the worker passed areas that should not be passed. It is easy to set individual or work specific dose limits. Several other system can be (and are already in process to be) connected to the EDOS, such as the whole body counting system, local dose information system and the local operating and maintenance system.
The introduction of a new electronic dosimeter also meant that it is now impossible for the workers to enter the controlled area without the dosimeter since it is needed to initiate the dosimeter when entering the controlled area. The isotropy of the EPD is more than 80% in all directions horizontally.

MEASURES TAKEN TO AVOID THE NON-PERCEPTION OF ALARMS FROM EPD CEI 61526

Javaraly FAZILEABASSE[1], Elisa CANAL, Nicolas VALENDRU[1], Patrice ROMANE[2]

[1] EDF DPN, 1, place Pleyel, 93200 Saint-Denis, France
[2] EDF CNPE Chinon, 37420 Avoine, France

Following events which took place in some PWR, and which were due to the lack of perception of alarms generated by EPD, EDF initiated research to answer the question: 'How can we guarantee to the agent the perception and knowledge of the chest EPD alarm- normalized according to the CEI 61526 – under any circumstances, in RCA?'

The difficulty of perception of the alarm is mainly due to extremely varied working conditions, such as:

- A very noisy environment: nature of the work, ventilation or turning engines, dismantling in progress, etc.
- Respiratory Protective Devices wearing
- Carrying the dosimeter in the breast pocket, and not near the eyes or ears.
- …

The research lead by EDF since 2010 is studying several aspects:

- An international 'benchmarking' through the ISOE, EPRI, WANO networks and work networks to benefit from both their REX and practice, and possibly to adopt the parry put in place to deal with this problem.
- A state of art about the supplementary alarm technologies- the EPD ones. Many other technologies exist – electrical shocks, thermal shocks-and are used in other areas to signal any danger.
- A study of the need which leaded to the creation of a wireless alarm -remote kit in addition to the EPD, and which must:
    - Enable an alarm transmission of the EPD Saphydose Gamma I (SGi)
    - Guarantee the knowledge of the alarm without disturbing the agent wearing the EPD in his activity, in:
        - Any type of working environment: very noisy, 'flashy', high effective Dose Rate, …, with or without the PPE (Personal Protective Equipment)
        - Any positions and situations which his work requires and without disturbing him in his activity.

In order to change nothing about the dosimeter, the kit will be composed of the following separate and distinct elements:

- An alarm emitter module,
- One or several alarm remote module(s) that can be worn at the same time,

This study leads to the realization of 5 kits in order to be approved in different activities. The test campaign is going to start as soon as January 2014.
The aim of the presentation during the meeting is to explain the solutions which were being used, the results obtained.

RADIATION PROTECTION IN DESIGN OF THE UK EPR

Author (1): Heather Hale, NNB - Radiological Protection Lead
Address: EDF Energy, Qube, 90 Whitfield Street, London, W1T 4EZ, UK
Phone: +44 7813 230196
Email: heather.hale@edf-energy.com

Author (2): Jean-Baptiste RICHARD, Radiation Protection Engineer
Address: EDF - Generation - Nuclear Department, 97 avenue Pierre Brossolette, 92120 Montrouge, Paris, FRANCE
Phone: +33 184226377
Email: jean-baptiste.richard@edf.fr

EDF are working on a project to build four units of the EPR in the UK. This project crosses the international boundary of the UK and France, by working together the EDF Energy's UK based Nuclear New Build (NNB) works with the EDF SA's Paris based EPR Design Engineering Department (CNEN).

This paper will demonstrate the benefits of designers & engineers working closely with operators to ensure good Radiological Protection, while explaining some of the specific challenges and technical information for the UK EPR, including:

- Dose optimisation choices & setting of the dose targets
- UK context modifications, including zoning & locked doors
- Developing an operational RP strategy to interface with the design engineering

PRACTICAL APPLICATION OF COMPUTER CODE PANTHERE FOR WORKERS' RADIATION PROTECTION

PIERRE CARLIER, XAVIER MICHOUX
Service Maintenance Levage / Groupe PR, EDF CIPN
140 Avenue Viton, 13009 Marseille - France
pierre-p.carlier@edf.fr

CLEMENT LERECULEY
CNPE Paluel, EDF
76450 Cany-Barville – France

ABSTRACT

The civil engineering operations to strengthen the raft of Fessenheim's nuclear plant were carried out by EDF. This technical modification has two principles objectives: 1) to increase the thickness of the reactor pit's concrete and 2) to create a new spreading area for corium (by creating a penetration through the wall of the reactor pit).

Behind the complex technical operations two radioprotection issues were studied using the computer program "PANTHERE": 1) Workers' radiation protection during the execution of the work (because of high dose rates in the reactor pit) and 2) operators' radiation protection after the execution of the work.

Results contributed to decrease personal and collective dosimetry of operations and to model and design a biological shield to protect workers during Fessenheim reactor operation.

1. Introduction

The civil engineering operations to strengthen the raft of Fessenheim's nuclear plant were carried out by EDF during the months of April to July 2013. This technical modification has two main objectives:

1. to increase the thickness of the reactor pit's concrete
2. to create a new spreading area for corium

These two main objectives respond to the request of the French Nuclear Safety Authority (ASN) to prevent corium from breaching the containment building and so to increase the overall safety level of the installation.

Behind the complex technical operations two radioprotection issues were studied:

1. Workers' radiation protection during the execution of the work. Dose rates in the reactor pit are high, from 2 mSv/h to 10 mSv/h.
2. But also workers' radiation protection after the execution of the work. Civil engineering operations also included a penetration through the wall of reactor pit that connects the spreading area and the reactor pit. This penetration may result in an increase of the dose rate in the spreading area when highly irradiating elements (such as thimble tubes) are in the reactor pit during routine maintenance and phases of unloading and reloading of reactor.

Firstly, studies were carried out with the computer program PANTHERE in order to estimate the impact of the position of the thimble tubes on the dose rate of reactor pit. The findings are the result of an optimum between nuclear safety and workers' radiation protection.

Then in a second stage, other studies using the same computer program were used to model and design a biological shield to avoid a potential radiological zoning change in the spreading area (due to the coring of the reactor pit wall). This biological protection is also the result of an optimum between a high nuclear safety level and workers' radiation protection.

2. Decreasing the dose rate of reactor pit

The reactor pit is the closed area which is located just under the reactor vessel. The main elements are thimble tubes which are present inside guide tubes. During operation of the reactor, this materiel helps to check neutron flux.

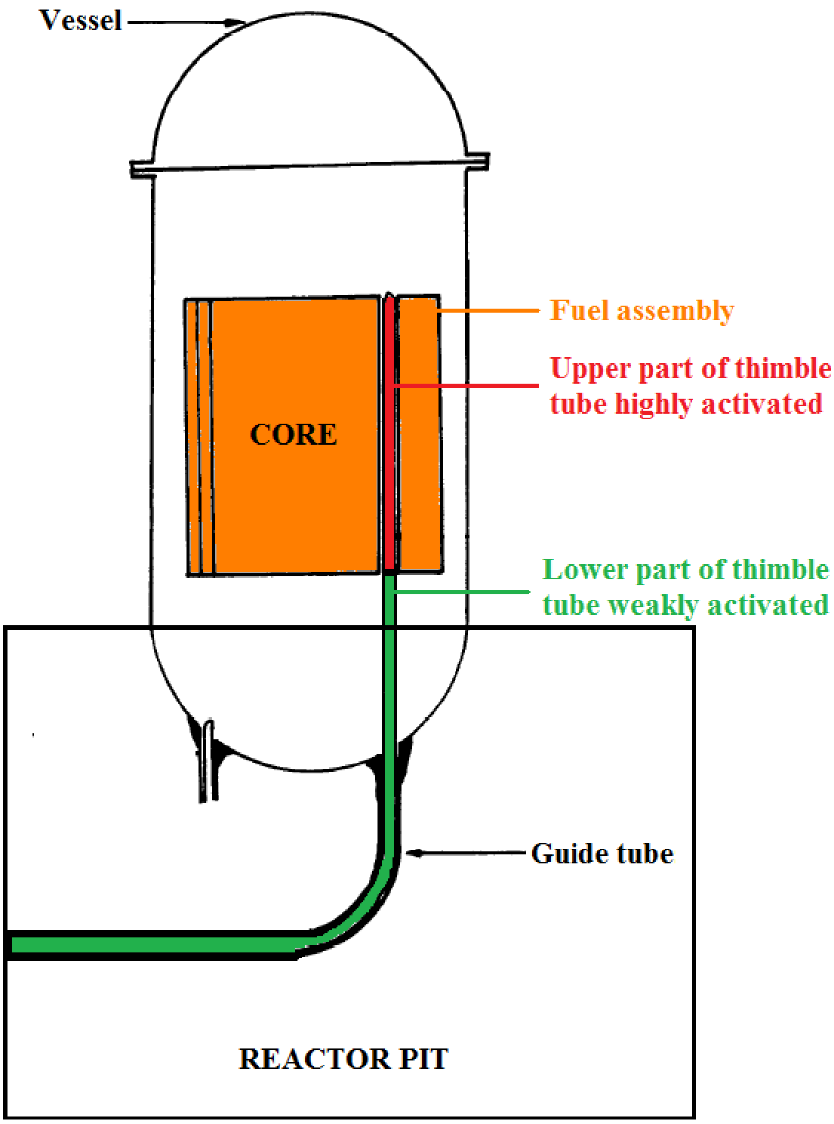
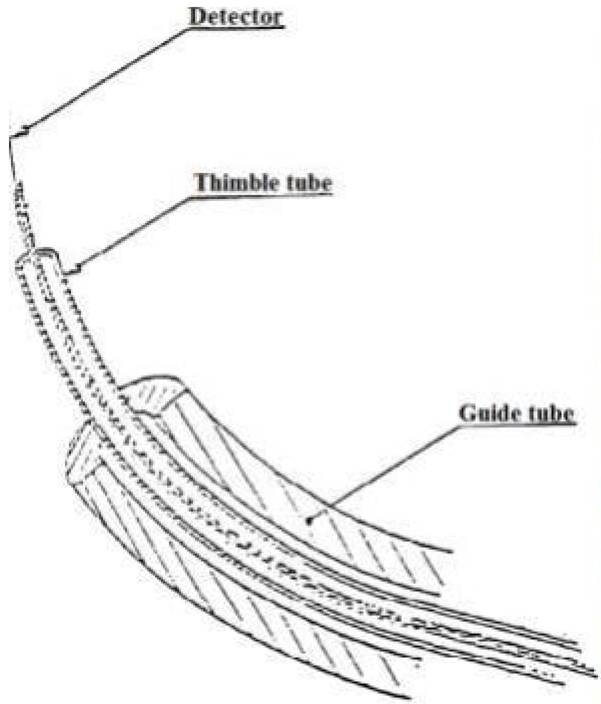


Figure 1: Overview



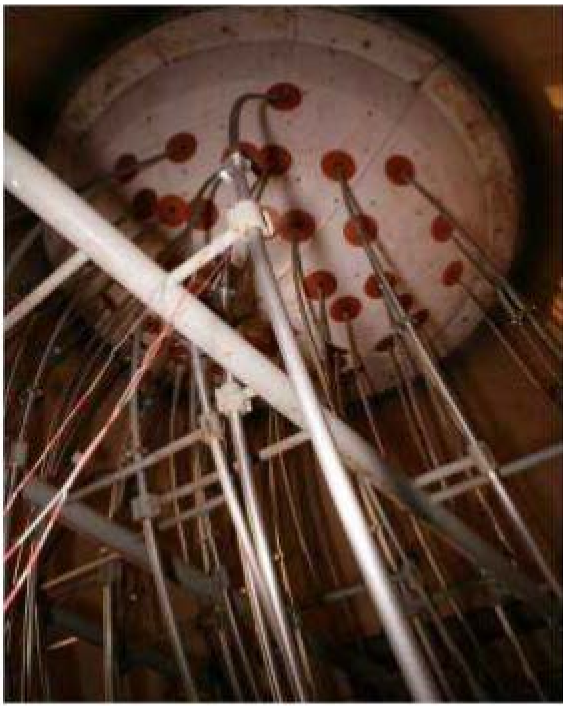


Figure 2: Guide tube pattern **Figure 3: Guide tubes in the reactor pit**

The guide tubes are fixed in the reactor pit while the thimble tubes can be inserted in the reactor core (when fuel assemblies are present) during operation and are retracted into the guide tubes when the reactor is stopped and defueled for maintenance. The latter involves an increase of the reactor pit's dose rate due to the activated portion (red part on figure 1) of thimble tubes being present in this area.

Usually, operations in the reactor pit are short and planned when the fuel is in the reactor vessel (reactor stopped). The fuel assemblies allow the thimble tubes to be guided inside the reactor vessel without which, there is a high risk of breaking the thimble tubes (which are very slender).

If we just take into account the radiation protection aspect, it is obvious that the best radiological conditions are reached when thimble tubes are fully inserted in reactor vessel. But it was not the safest for the integrity of thimble tubes because of the constraints of planning as the operations in the reactor pit must be done while the fuel is unloaded (thimble tubes wouldn't be supported).

A compromise between radiation protection and thimble tubes position was reached thanks to the computer program "Panthere". First, the reactor vessel, the reactor pit, upper part of the thimble tubes (which are strongly activated) and the guide tubes were modeled. Then, reference dose rates were computed (and checked with in situ measurements) in the case where the thimble tubes are fully inserted into the reactor vessel (best configuration for radiation protection).

	In situ measurements	With fully inserted modeling
Reference dose rate (in mSv/h)	Between 1.7 and 3 mSv/h	2.4 mSv/h (total average on 25 measurement points)

Tableau 1: Reference dose rate

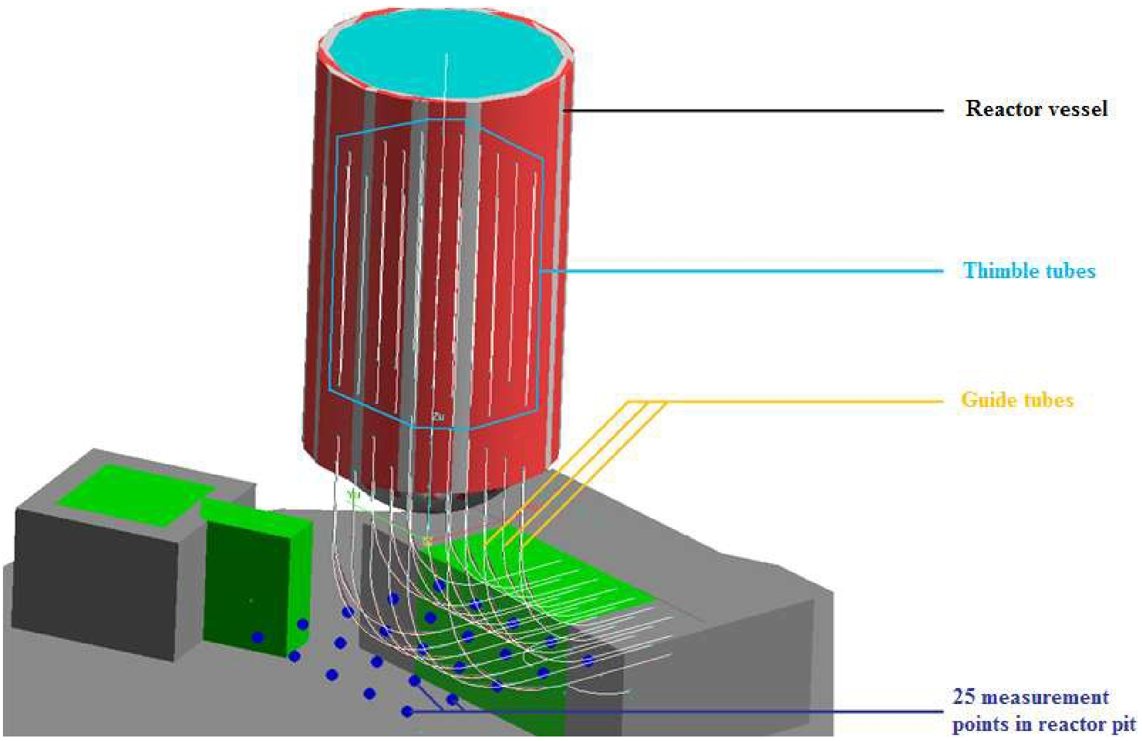


Figure 4: Overview of model (Panthere)

Then a parametric study was performed in order to evaluate the dose rate's evolution in the reactor pit as a function of the position of the strongly activated upper part (red part in figure 1) of the thimble tubes from the bottom of the vessel.

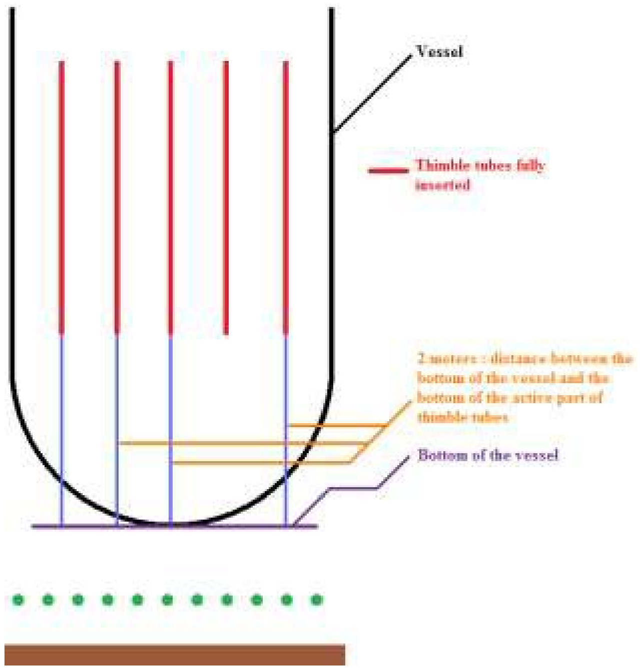


Figure 5: Thimble tubes fully inserted

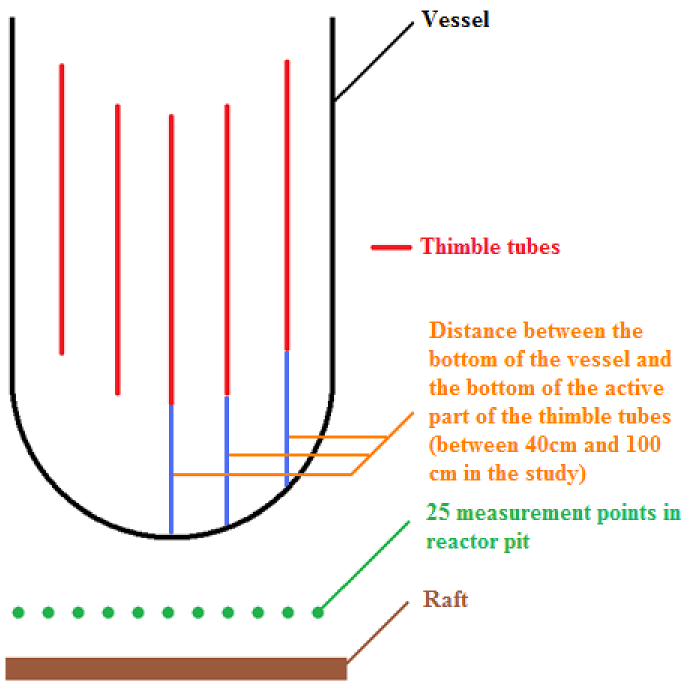


Figure 6: Thimble tubes partially inserted

Distances which were studied range from 40cm to 100cm. As expected, the further the thimble tubes are far from the bottom of the vessel, the lower the dose rate is.

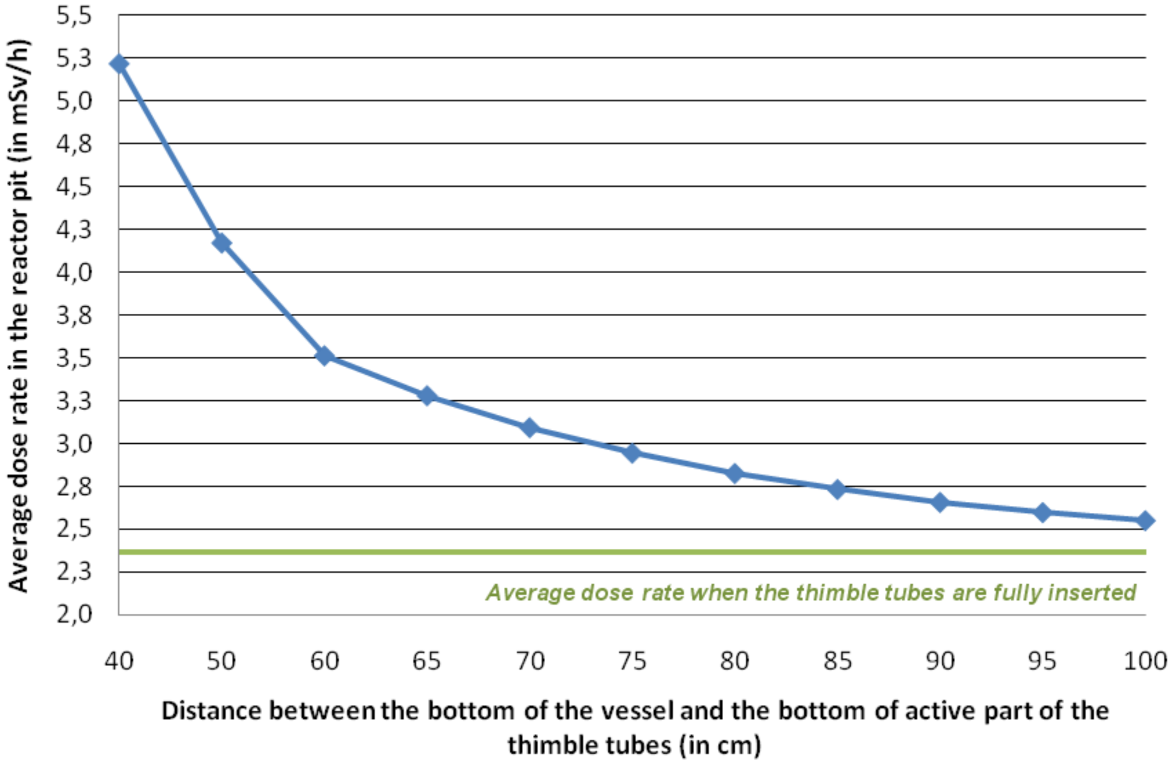


Figure 7: Average dose rate in the reactor pit as a function of position of the thimble tubes

	Distance between the bottom of the vessel and the bottom of active part of the thimble tubes										
	40	50	60	65	70	75	80	85	90	95	100
Average dose rate in the reactor pit (in mSv/h)	5,2	4.2	3.5	3.3	3.1	2.9	2.8	2.7	2.7	2.6	2.6

Tableau 2: Average dose rate in the reactor pit as a function of position of the thimble tubes

These results show that for distance range from 85 to 100cm, the average dose rate is close to the reference dose rate (difference is less than 15%).

The findings of this study were shared with Fessenheim's nuclear plant and applied during the operations in the reactor pit. This strongly contributed to decrease personal and collective dosimetry.

3. Designing and modeling the biological shield

One of the main steps of the civil engineering operations was to create a new spreading area for corium by coring the wall of the reactor pit. It consisted of removing 1.8 meter thickness of concrete and it has an effect on the dose rate of adjacent spaces, especially when the strongly activated upper part of the thimble tubes are stored in the reactor pit during routine maintenance and phases of unloading and reloading of reactor.

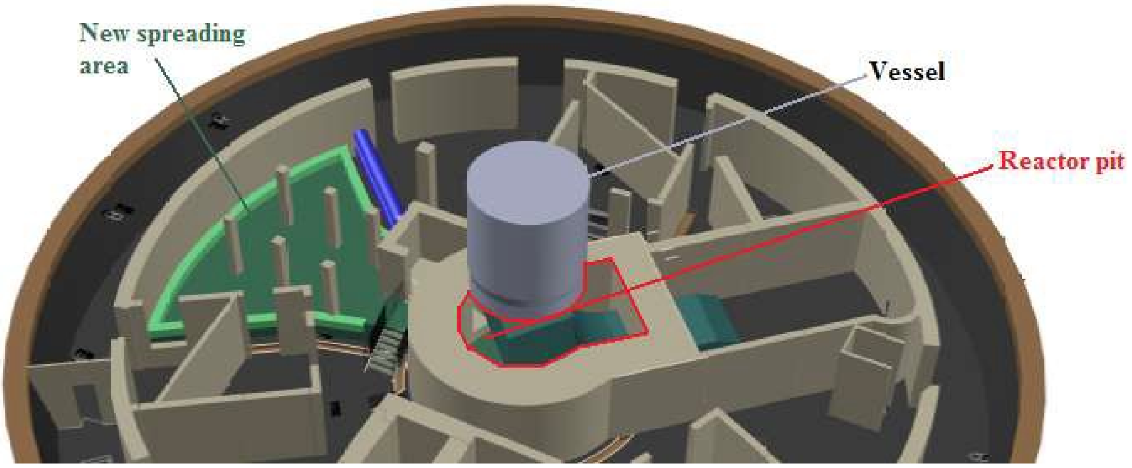


Figure 8: Overview of reactor pit and new spreading area

An increase of the dose rate in this area is not acceptable for the plant operator because it leads to an increase in personal and collective dosimetry. Furthermore access to these areas of high dose rate is strictly controlled.

In order to avoid a potential radiological zoning change, a biological shield was modeled and designed with the help of the computer program Panthere. The first calculations were run without the coring in the aim to establish the reference radiological zoning. Then the coring was added to the numeric modeling to obtain the radiological zoning with the impact of the coring.

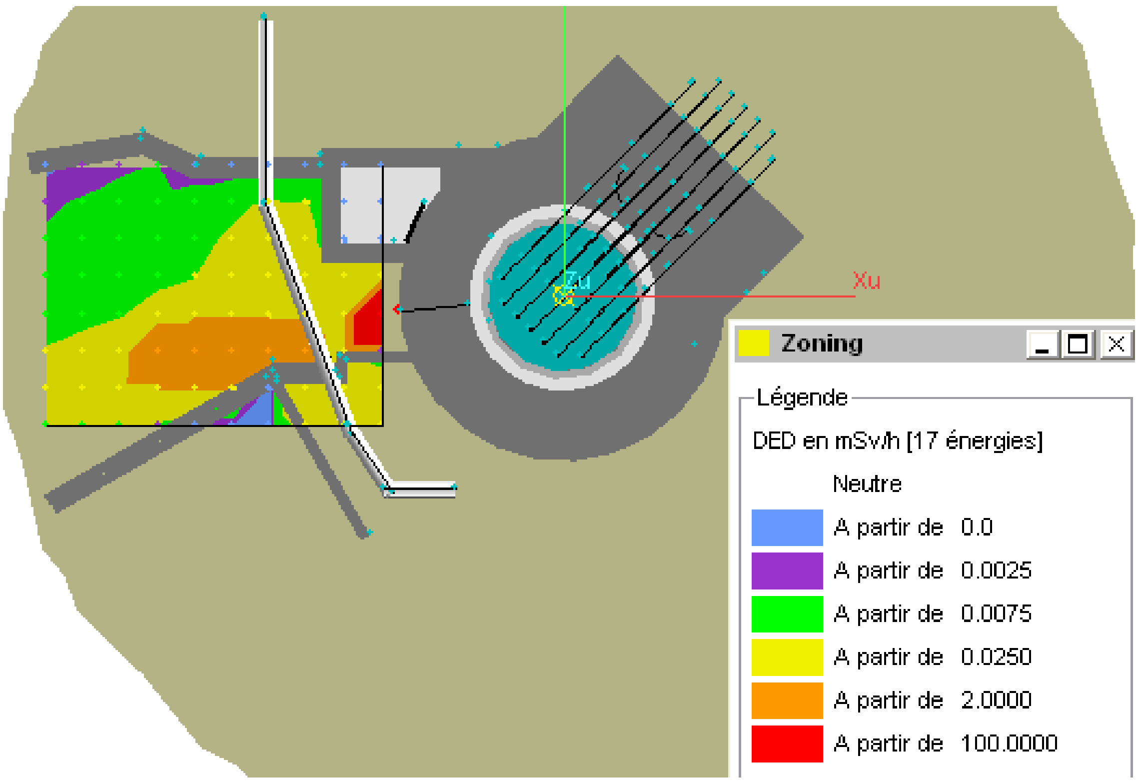


Figure 9: Radiological zoning due to the coring

With these first results the biological shield was designed with the followings technical requirements:

- No prohibited area (dose rate > 100 mSv/h, in red in figure 9)
- No limited stay area (2 mSv/h < dose rate < 100 mSv/h in orange in figure 9) beyond 50 centimeters after the coring
- Biological shield shall be implanted in the reactor pit and shall not block the coring through the wall of reactor pit.

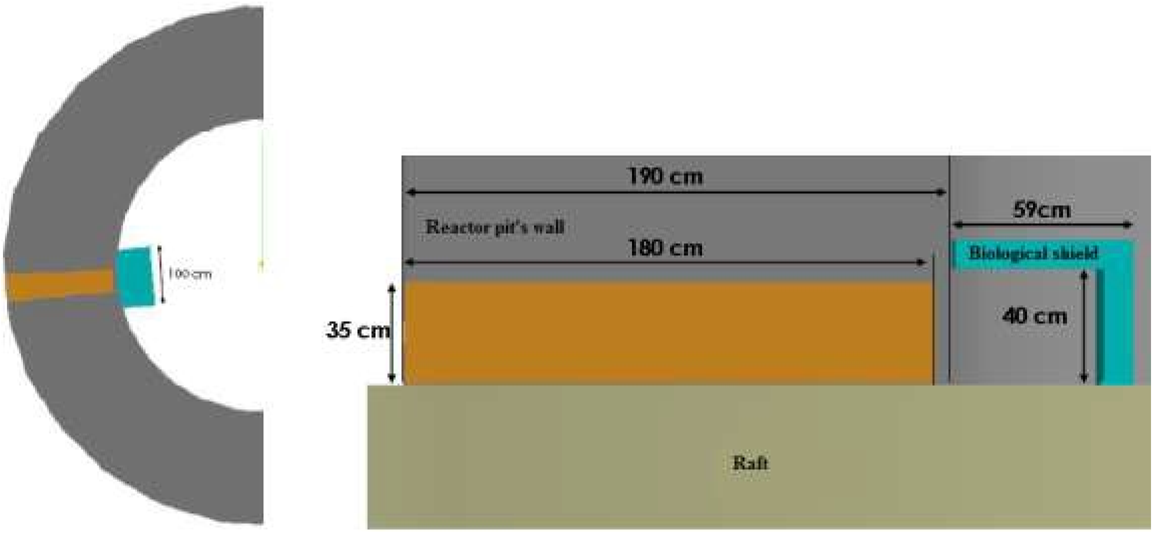


Figure 10: Biological shield's pattern

The L-shape allows corium flow while reducing radiation (no direct radiation through the coring). After several iterations of Panthere, the above dose rate objectives were reached with an 11 centimeter thickness of lead biological shield.

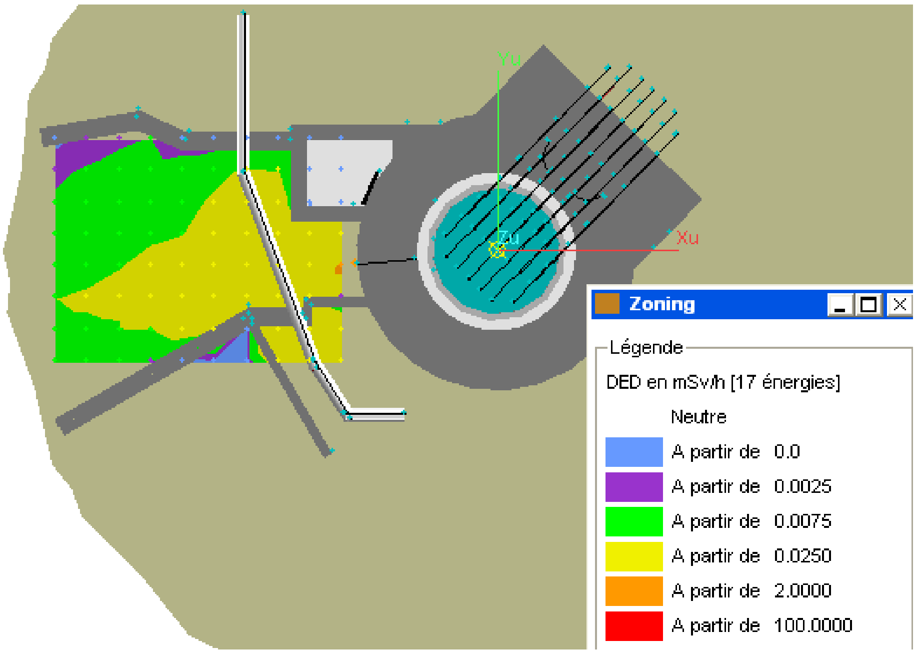


Figure 11: Radiological zoning with biological shield

Radiological zoning on figure 11 shows two “orange” points: one at the exit of the coring and the other just after 50 centimeters (3.3 and 2.6 mSv/h respectively). Apart from these two points, the spreading area is a regulated stay area (0.025 mSv/h < dose rate < 2 mSv/h in yellow in figure 11).

After the end of the operations, a measure was made in order to check the correct designing of this shield. The in situ measure gave 0.5 mSv/h at the exit of the coring (compared to 3.3 mSv/h with modeling).

Figure 12: Biological shield in the reactor pit

4. Conclusion

The results of the first study have been largely validated by measurements made in real time during different steps throughout the execution of the work. This has, largely, allowed the dosimetric objectives defined in the ALARA approach to be reached.

The second study included a look at the future operational needs of the nuclear plant operator. The many discussions with the nuclear plant helped to define the fundamental requirements so that the engineering operations have the least impact on the radiological zoning. Thanks to the biological protection, the creation of the coring through the wall of reactor pit has only a minor impact on the dose rate of the spreading area.

In conclusion, these studies show the importance of taking into account the radiation protection from the design phase of solutions, to execution of the technical operations and through the future needs of the plant operator.

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CONTAINMENT ELECTRICAL PENETRATION ASSEMBLIES AND CABLE REPLACEMENT IN UNIT 3 NPP OKG SWEDEN

Robin Hanzl
Manager, Environment, Radiation Protection

OKG Aktiebolag
SE-572 83 Oskarshamn

Tel +46 491-78 67 36 - Mobile +46 70- 603 88 78 - Fax +46 491-78 60 90
robin.hanzl@okg.eon.se

During the outage year 2014, OKG will implement a project called MILK (Containment electrical penetration assemblies and cable replacement). The time schedule is for the outage 55 days and for project MILK the schedule is 40 days including installation and testing. The project target is that all cables that connect instruments, motors, valves inside of the containment with the connecting equipment at the outside of the containment will be replaced in this time.

The work has been estimated to cover around 15 000 working hours inside the containment. Totally amount of people inside the containment is number to

To reduce collective- and individual dose has a number of steps been taken and proposed. The goal is to go from an estimated collective dose of 1.7 manSv to an estimated collective dose of 0.5 manSv.

To gain the goal of dose, the project has

- made use of the experience gained from similar projects
- used a method of laser scanning of the environment in the containment in order to simulate the cable routes
- the database of laser scanning is also used in order to fabricate the new cables
- proposed a decontamination of the reactor's cooling systems
- prefabrication of cables
- optimize components in premanufacturing as much as possible avoiding conventional installation work
- planned the logistics
- introduced a new way to deal with installation instructions by using electronic document handling
- training of staff in mock up facility
- trained staff and format them into specialized teams where everyone knows there own tasks

CERNUM: AN ADVANCED TRAINING TOOL IN THE SERVICE OF RADIATION PROTECTION 2013 FIRST APPLICATIONS: SIMULATION OF IRRADIATING SOURCES

E. COURAGEOT, E. GAILLARD-LECANU, R. KUTSCHERA, G. LE MEUR
R&D/STEP, EDF
6 Quai Watier, 78400 Chatou – France

A. RIEDEL
DPN/UNIE/GPRE, EDF
1 place Pleyel, 93200 Saint-Denis, France – France

B. THERACHE, O. MENARD
CNPE Civaux, EDF
Centrale de Civaux, BP 64, 86 320 Civaux – France

Any personnel involved in activities within the controlled area of a nuclear facility must be provided with appropriate radiological protection training. An evident purpose of this training is to know the regulation dedicated to workplaces where ionizing radiation may be present, in order to properly carry out the radiation monitoring, to use suitable protective equipments and to behave correctly if unexpected working conditions happen.

A major difficulty of this training consists in having the most realistic reading from the monitoring devices for a given exposure situation, but without using real radioactive sources. EDF R&D is developing a new approach of radiological protection training called CERNUM (French for “numerical radiological protection training”). Combining different technologies, into an environment representative of the workplace, but still geographically separated from the nuclear power plant, the CERNUM is build upon: a training area representative of a workplace, a geolocalization system, a Man Machine Interface used by the trainer to define the source configuration and the training scenario, fictive radiation monitoring devices and a particle transport code able to calculate in real time the dose map due to the virtual sources placed in the simulated area.

CERNUM is deployed on the French nuclear site of Civaux. Its first training feedbacks are very positive. The system allows simulation of both hot spots and ambient dose rate. For example, a 2µSv/h ambient dose equivalent rate can be simulated as a single ^{60}Co source with a 7 GBq activity which has been virtually placed 10m above the roofless virtual training area. Using the fictive radiameter, the trainee could successfully measure the 2µSv/h ambient dose equivalent rate. Additionally, one scenario involves a hotspot with a 70GBq activity placed inside a pipe, resulting to a 2mSv/h dose equivalent rate at the worksite in close proximity. Based on localization of the worker undergoing training, the fictive dosimeter and radiameter allowed determination of the dose equivalent and the ambient dose equivalent rate in the vicinity. Using the radiameter, the worker could track the hotspot within a 15 cm positioning error, due to the localization hardware’s limitations.

These are just mere examples training scenarios achievable using CERNUM as an efficient solution for a realistic training. EDF is still working on more use cases, ranged from basic radiological training to advanced experts courses.

**DECONTAMINATION OF THE 4 STEAM GENERATORS, LOOP LINES AND PRESSURIZER FOR DECOMISSIONING -
EXPERIENCE AT THE NPP CHOOZ A IN FRANCE**

Main Author: Reymann, Dr. Sébastien; Moreira do Amaral, Pedro
Co-Author: Neuhaus, Dr. Rainer; Sempere-Belda, Luis; Basu, Ashim

AREVA GmbH, Paul-Gossen-Str. 100, 91052 Erlangen, Germany,
sebastien.reymann@areva.com , jose_pedro.moreira_do_amaral@areva.com ,
luis.sempere-belda@areva.com , rainer.neuhaus@areva.com , ashim.basu@areva.com

Chooz A, was the first PWR built in France - a four loop plant delivering 307 MWe. The plant was in operation from 1967 until 1993. Chooz A NPP is also the first PWR being decommissioned in France. The first priority of the operator EDF was to dispose large components with the highest contamination and activity as very low level waste. This task was the first decontamination project of AREVA GmbH in France.

Due to the technical requirements EDF decided to decontaminate the large components one after the other. The four steam generators were removed from their original position and laid down horizontally before being decontaminated. The pressurizer and loop piping were decontaminated in-situ. The decontamination operations performed by AREVA were integrated in the decommissioning schedule. The decommissioning was performed by ONET Technologies Nuclear Decommissioning.

In this presentation we will describe the decontamination application and results. The four steam generators, the pressurizer and the loop piping were decontaminated in seven separate decontamination operations. Due to the composition of the oxide layer, an adaptation of AREVA's well-known CORD UV® process had to be applied to achieve the decontamination results as defined in the specification.

CONDITIONING OF INTERMEDIATE LEVEL RESIN BY DEWATERING: A NOVEL EXPERIENCE FOR THE UK

Matt Lunn
Radioactive Waste Adviser
EDF Energy
Sizewell B Power Station
Leiston
Suffolk, IP16 4UR
United Kingdom

Tel: +44 1728 653750
Fax: +44 1728 653189
E-mail: matthew.lunn@edf-energy.com

In the UK, conditioning of ion-exchange bed resins for disposal as either low or intermediate level waste has traditionally been based around cement encapsulation in thin-walled stainless steel drums.

Independently, both EDF Energy and Magnox have proposed the use of ductile cast iron containers for the storage, transport and disposal of de-watered resins and sludge, at both operating and decommissioning nuclear power plants.

Introduction of the ductile cast iron container and the dewatering waste conditioning process represents a major shift in the UK's waste disposal practices.

This paper describes our experiences with introducing a novel container type and waste conditioning process, and it will discuss the design and implementation of the plant modifications necessary to accommodate the resin dewatering plant. This paper will also provide our operational feedback with respect to conditioning 25m^3 of ILW bead resins into 55 MOSAIK II-15 casks, using the FAFNIR V and NEWA, supplied by GNS.

ALARA MANAGEMENT MEASURES AND EXPERIENCE IN POST HANDLING OF REPLACED PRESSURIZER FROM RINGHALS 4

Eric Hernvall & Torgny Svedberg
Ringhals NPP, Vattenfall, Sweden

In 2011 Ringhals 4, Westinghouse PWR 3 loop reactor, replaced Steam Generator (SG) and Pressurizer (PRZ). This was the latest SG replacement at Ringhals. All three PWR at Ringhals has replaced its Steam Generators.

Because of issues concerning the material integrity in the spray and surge line nozzles, the heaters, heater penetrations and issues in other repair cases; Ringhals saw the unique opportunity to disassemble the pressurizer to perform deeper material investigations, after 27 years of operation.

The work on the pressurizer started in fall 2011 and has proceeded into 2013.

An ALARA plan was initiated early in the work planning process in order to optimize the routings concerning radiation protection implementation in the work scoop.

In order to perform an accurate and thoughtful ALARA optimization, a number of radiation protection measures have been taken throughout the work.

Examples on ALARA implementation can be, preparing work shop for handling PRZ and radioactive material, source term management, work logistics, shielding calculations and implementation, optimization of specialized tools concerning optimization of person dose and spread of contamination, contamination control, decontamination etc.

Today, in okt 2013, the pressurizer is practically totally dismantled and the cut off upper and lower calottes will be used in future as mock up.

DESIGN OF TRANSPORT CONTAINER FOR THE REDUCTION OF OCCUPATIONAL EXPOSURE

Erwin Neukäter
Head of Radioactive Waste
BKW FMB Energy Ltd.
Nuclear Power Plant Mühleberg
Kraftwerkstr.
CH-3203 Mühleberg
erwin.neukaeter@bkw.ch

Until now KKM produced 4'500 drums of cemented raisins and radioactive waste deriving from the reactor (spent fuel channels etc.) which are stored on site at KKM. In preparation for decommissioning of KKM this storage space is needed. Therefore all drums have to be transported by road to an intermediate storage facility as a special geological storage is not yet available in Switzerland.

Since some of the drums have dose rates of several Sv/h some means of shielding is needed during the transport in order to meet the legal transport requirements.

In order to transport the 4'500 drums in a timeframe of four years and to save personal dose for personal involved in the transports a special transport container was designed and built. This special 10'-container (hight 1.3m, overall weight <12Mg) is capable of carrying five drums with a special shielding of the inner position (top, bottom and sides). Transport of three containers simultaneously with a max. load of 40 Mg.

PRIMARY SYSTEM ACTIVITY AND IMPACT ON RADIATION FIELDS

David Perkins
Program Manager

EPRI (Electric Power Research Institute)
3420 Hillview Avenue
Palo Alto, CA 94304 U.S.A.
(817-691-6494)
dperkins@epri.com

Radiation fields are developed over time from the deposition of activated corrosion products into primary surfaces. In most cases, these fields are dominated by the deposition of 58Co and 60Co on out-of-core surfaces. The impact of coolant activities on shutdown dose rates varies depending on several factors. Using a simplistic model and approach, it is estimated that cycle 58Co surface activities are impacted more over the last 3 – 6 months of operation, while the surface activity of 60Co is developed over several years before reaching equilibrium.

The replacement of major component replacements can have an impact on radiation fields due to the introduction of new materials or source term for activation. Industry data following steam generator replacement has shown initially higher dose rates followed by a lowering dose rate trends after the second or third cycle of operation.

Source term is impacted by many factors. These factors can be impacted by materials, chemistry controls, fuel design and plant materials. Each of these factors has an impact on coolant activity and present challenges to radiation protection, chemistry and engineering personnel.

This presentation will provide an overview of the factors impacting coolant activity and current observations based on current industry operating experience and application of the basic model related to plant dose rates.

SOURCE TERM REDUCTION VIA WATER CHEMISTRY AS ONE PART OF THE INTEGRATED REGULATORY SUPERVISION OF RADIATION PROTECTION IN SWISS NUCLEAR FACILITIES

H. Glasbrenner
Swiss Federal Nuclear Safety Inspectorate
(Eidgenössisches Nuklearsicherheitsinspektorat, ENSI)
Heike.Glasbrenner@ensi.ch

The division operational radiation protection within the Swiss Federal Nuclear Safety Inspectorate (ENSI) also supervises water chemistry in Swiss nuclear power plants. In this talk some practical links between water chemistry and radiation protection will be shown.

It is well known, that there are several reasons why water chemistry in NPPs is relevant and has to be controlled to make sure that different parameters and concentrations of some special nuclides not exceed the limit values limits proposed. Good water chemistry can guarantee the integrity of the fuel rods and mitigate different kinds of corrosion like PWSCC (primary water stress corrosion cracking) and erosion corrosion. Little corrosion leads to little corrosion products which can be activated and can contribute to the dose rate. Hence it is obvious that water chemistry plays an important role by reducing the dose rates in power plants and protects the people working on site.

In this presentation it is shown by relevant examples generated in Swiss power plants how changes in water chemistry can influence dose reduction.

INVESTIGATIONS INTO SURFACE TREATMENT METHODS FOR REDUCTION OF RECONTAMINATION OF BWR REACTOR SYSTEMS

Bernhard Stellwag [1)], Juan de Dios Sanchez Zapata [2)], Ashim Basu [1)], Stephan Hoffmann-Wankerl [1)], Luis Sempere Belda [1)], Tomas Berasaluce [2)]

[1)] AREVA GmbH, Erlangen, Germany
[2)] Iberdrola Generation S.A.U., C.N. Cofrentes, Cofrentes (Valencia), Spain

The NPP Cofrentes is a 1100 MWe-class BWR plant designed by General Electric (GE reactor generation BWR 6). 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 (OLNCTM: on-line noble metal chemical addition, the process is patented by General Electric). In September 2013, the NPP Cofrentes is in its 19th refueling outage (RFO 19). 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). The plant operator therefore considers another decontamination of the RWCU system in 2015 (RFO 20). The decision will be made depending on the dose rate level at the system in the outage this year and on the work scope planned for the outage in 2015.

Given this background, the plant operator of NPP Cofrentes and AREVA started a joint R&D program, this year. It 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 LTNCTM, the process is patented by General Electric); (2) Application of a selfassembling 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 will be tested by an in-plant exposure program at NPP Cofrentes. The treated tube samples will be installed in the mitigation monitoring system (MMS) of the plant, which allows the exposure of these samples to the reactor water under operating conditions. The first set of samples will be removed three months after start-up of the plant to its 20th fuel cycle, i.e. in spring 2014.

Results are reported in this contribution.

^{60}CO CONTAMINATION OVERVIEW ON EDF FLEET

G. Ranchoux [1], M. Benfarah [1], F. Drouet [2], F. Dacquait [3], L. Guinard [1]

[1] EDF/SEPTEN, 12-14 avenue DUTRIEVOZ, 69628 Villeurbanne, France, Tel: +33 4 72 82 77 67, gilles.ranchoux@edf.fr
[2] EDF/DPN/UNIE, [3] CEA/Cadarache

Dose rate reduction during outages is a strategic purpose for EDF (working conditions improvement during outage, nuclear acceptability and respect of regulation). The reach of these main purposes is correlated to source term reduction and, more specifically to ^{60}Co contamination management in PWRs primary and auxiliary circuits.

This paper will give a state of the art in ^{60}Co contamination mechanisms (origin and behavior in primary circuit). A review of the ^{60}Co contamination (volume activity, deposited activity and contribution to dose rates) on French fleet will also be detailed and compared to the international feedback.

Moreover, a survey dedicated to ^{60}Co hot spots contamination was rolled out in 2013 at EDF in order to determine their usual location and their main causes (natural material release or wear, incident on equipments, bad maintenance practices). The main results of this survey will be presented in this paper.

Finally, this article will present the prospects that will lead EDF to reduce preventively “at the root” the ^{60}Co contamination impact on radiation protection of workers in the future.

HIGH PRESSURE WATER DECONTAMINATION AT THE NUCLEAR POWER PLANT PHILIPPSBURG

Michael Hellmann

EnBW Kernkraft GmbH
Kernkraftwerk Philippsburg
Rheinschanzinsel
76661 Philippsburg
m.hellmann@kk.enbw.com

High pressure water decontamination at a nuclear power plant is used to reduce dose rate and contamination by removing radioactive deposits from inner surfaces of tubes, vessels an heat exchangers.

The presentation shows some applications, used equipment and dose analyses.

EXELON'S CAREFUL AND CRITICAL COMPARISON OF TWO SOURCE TERM REDUCTION METHODS IMPLEMENTED AT BYRON-1 AND BRAIDWOOD-1

Rick Leasure, Exelon Braidwood Radiation Protection Manager
Jamison Rappeport, Exelon Braidwood Chemistry Manager
Harry Miller, Exelon Braidwood, ALARA
PJ Robinson, (n,p) Energy, Inc.
rick.leasure@exeloncorp.com

Abstract

Exelon Braidwood-1 has reduced radiation dose rates at a very rapid rate, -75% decline in lower containment dose rates, in 3 years, 3 refueling outages. The Braidwood performance improvement has shifted the INPO/WANO ranking from 4th Quartile to 1st Quartile. The "code has been cracked" in how to reduce source term at PWRs, fast and cost effectively at PWRs. The purpose of this paper is to share with the International RP community, the results, from a careful and critical comparison of two different methods. This comparison was conducted at two Exelon PWRs with identical designs to enable the determination of the best available solution for source term for Exelon, which resolves a long standing technical controversary. The nuclear power industry has many choices in technology to try to reduce plant dose rates including: zinc injection, fuel cleaning, specialty resins, RCS filtration, full or partial system decon and cycle chemistry. The challenge for the RPM, and site organization, is to select which of the technologies produces: 1) the maximum reduction in dose rates at a reasonable cost, 2) a return on investment in shortest time which supports outage overall performance and 3) provides consistent sustainable results. This paper presents the approach, technology use, and results which accelerated the Braidwood-1 decline in dose rates, such that, their Sister Unit Byron-1, has dose rates that are 2 times higher. In the future, Braidwood will be completing 10 to 15 REM, refueling outages by 2015.

Background

For the past 10 years, in the United States, and within Exelon, there was considerable technical controversary of two methods being used to reduce radiation source term within the chemistry organizations and EPRI. The methods involved two very different approaches and two very different technologies. At Exelon, the Corporate Senior Leadership (SLT) team, lead by the Chief Operating Officer, has a strong commitment to new and innovative solutions for ALARA, and initiatives to reduce station dose rates, which reduces station occupational radiation exposure. In 2010, the Corporate SLT and Exelon Board of Directors, Oversight Committee, received reports of a new technology to reduce radiation levels demonstrated successful at other utilities, but was not yet implemented at Exelon PWRs. Exelon Station RP organization had identified the technology, as it emerged in 2003. However, the new technology and related sequence for reactor shutdown and start-up, was not implemented due to a significant technical controversy that existed with station and industry chemists. The new technology is proprietary, and the only way to fully protect the intellectual property and small business concerns, was to not disclose details of function. The Exelon COO identified and directed an innovative plan to vett, or to conduct a careful and critical comparison of the two technical solutions: an industry/Exelon standard solution, and the new technology solution, and their effectiveness in reducing radiation source term. The only way to confirm efficacy was to do a critical comparison of the two approaches over three refueling outages between sister units. The rate of change radiation source term was compared for Byron-1 and Braidwood-1; and Byron-2 and Braidwood-2. The Corporate and Station SLT approved the project that established Braidwood-1,2 as the lead Exelon PWR for integration of new technology to reduce plant radiation levels. Further, Exelon Corporate SLT, provided the nuclear power industry with final resolution of an industry technical controversy.

REACTOR CAVITY DECONTAMINATION IMPROVEMENTS AT EDF

Marie Michelet[1], Serge Blond[2]

[1]EDF/DPN/UNIE/GPRE, Cap Ampère, 1 Place Pleyel, 93282 Saint Denis Cedex marie.michelet@edf.fr, 01 43 69 44 72

[2] EDF/DPN/EM RP, Cap Ampère, 1 Place Pleyel, 93282 Saint Denis Cedex

Recent surveys on reactor cavity decontamination at EDF NPPs have pointed out the following:

- Heterogeneous process in the fleet
- The standard process is sometimes not enough sufficient to reach the radiological objectives
- Several events in terms of contamination, etc.

In order to improve practices a working group has been set up to discuss current practices and homogenize the reactor cavity decontamination at EDF. In parallel, an EPRI-EDF workshop is organized at the end of September to exchange about international practices. The ISOE network has contributed to this workshop through a questionnaire launched at spring 2013 in order to collect and compare the utilities practices.

The objective of the presentation is to expose the conclusions of the EDF working group, the future work on this topic at EDF and the main conclusions of the EDF-EPRI Workshop.

RADIATION PROTECTION ASPECTS OF WATER CHEMISTRY AND SOURCE-TERM MANAGEMENT

A. Rocher, G. Ranchoux, G. Abela (EDF, France), L. Vaillant (CEPN, France), D. Wells, D. Perkins (EPRI, USA), M. Olsson (Vattenfall, Sweden), I. Smieško (ENEL, Slovak Republic), D. Miller (NATC, USA), W. Harris (Exelon, USA), H.B. Okyar (OECD/NEA, France)

Water chemistry approaches in different design of NPPs vary in results and consequences in terms of radiation protection performance. The publication primarily focuses on three topics dealing with water chemistry, source term management and remediation techniques. For each approaches, it is expected to identify how radiation protection benefits are evaluated with a focus on measurement techniques such as CZT gamma spectroscopy. One key objective of the report is to provide current knowledge regarding these topics and to address clearly related radiation protection issues. In that mind, the report prepared by the Expert Group Water Chemistry was also reviewed by radiation protection experts. The outcome of the work is a new ISOE publication. Various designs PWRs, VVERs, and BWRs are addressed within the document.

EQUIVALENT DOSE TO THE LENS OF THE EYE AT NUCLEAR INDUSTRIAL WORKPLACES AND SHIELDING FACTORS FOR PROTECTIVE EYE WEAR – MEASUREMENTS AND CALCULATIONS

Lisa Bäckström, Karin Andgren

Vattenfall Research & Development AB
Business Division Nuclear Power
Box 475, SE-401 27 Göteborg

lisa.backstrom@vattenfall.com and karin.andgren@vattenfall.com

The project:
During the second half of 2013, eight (8) Swedish nuclear industrial workplaces together performed a study regarding equivalent dose to the lens of the eye. The working group consisted of representatives from the eight sites and Vattenfall Research and Development AB, VRD was holding the project together. Among the participating sites were nuclear power plants, a nuclear fuel factory and waste workplaces.
The project was a result of requirements from the Swedish Radiation Safety Authority, and the outcome of the project was reported to the authority in January 2014.

Study:
Calculations of equivalent dose to the lens of the eye
The focus of the study was to identify situations/professions where the whole body dosemeter is not god enough for estimation of the eye lens dose. For most sites this means, identifying when measurements of Hp(10) and Hp(0,07) at the chest is not god enough for estimation of eye lens dose. Critical situations are when the whole body dosemeter is shielded and/or when beta particles with energies above 700 keV are present.
The sites identified such situations and equivalent dose to the lens of the eye was estimated using MicroShield (for photon radiation) and MCNP5 (for beta radiation).

Testing of headband dosemeter measuring Hp(3)
Reference irradiations for a TLD headband dosemeter from Public Health England, PHE was performed at Physikalisch-Technische Bundesanstalt, PTB in Germany. The purpose of the testing was to study the fulfilment of performance requirement for personal dosemeters set up by the Swedish Radiation Safety Authority.

Measurements of equivalent dose to the lens of the eye
During October and November 2013 measurements of the equivalent dose to the lens of the eye in terms of Hp(3) were performed. 140 workers used the headband dosemeter from PHE during their work at radiological controlled area. The doses registered by the headband dosemeters were compared to the doses registered by the whole body dosemeters.

Shielding factors for protective eye eyewear
Shielding factors for eight (8) different protective eyewear were studied. The tested shields were; protective eye glasses, full face mask, mururoa, procap and other visors. Irradiations with both photons and beta particles were performed at PTB and the shielding factors were estimated, using MicroShield (for photon radiation) and Varskin (for beta radiation).

NEW MONITORING SYSTEM TO DETECT A RADIOACTIVE MATERIAL IN MOTION

Karim BOUDERGUI, Vladimir KONDRASOVS, Romain COULON, Gwenolé CORRE, Stéphane NORMAND

CEA, LIST, Sensors & Electronic Architectures Laboratory, F-91191 Gif-sur-Yvette CEDEX, FRANCE

Radioactive material transportation detection is important especially in nuclear power plant.
Industrial systems include Radiation Portal Monitor (RPM) to detect radioactive matters transported in vehicles or carried by pedestrians. However, today's RPMs are not able to efficiently detect a radioactive material in movement. Indeed, integration time of several seconds or minutes requires stopping the device to scan.

As count statistic may be low depending on the nature of the radioactive material and its packaging, long integration time is required to differentiate gamma background from the actual signal emitted. It implies that false alarms may be triggered or at the opposite a radioactive material not detected. This last scenario is due to a lack of count statistics, mainly because of the vehicle speed that is too fast in respect with the integration time. The statistical false alarm rate has to be as low as possible.

The real-time approach depicted in this paper consists in using a time correlated detection technique in association with a sensor network. It is based on several low-cost plastic scintillators and a digital signal processing designed for signal reconstruction from the sensor network. The number of the sensors used in the network can be adapted to fit with applications requirements (efficiency …) or cost. The reconstructed signal is improved by comparing to other approaches. This allows us to increase the device speed that has to be scanned while decreasing the risk of false alarm.

This paper presents the method developed in our laboratory and results obtained during an experiment in Milan urban mass transportation in the framework of a project so called Secured Urban Transportation - European Demonstration SECUR-ED.

AN INVESTIGATION OF THE DISTRIBUTION OF ALPHA AIRBORNE RADIOACTIVITY AT FORSMARK NPP

Lina Ekerljung (lina.ekerljung@forsmark.vattenfall.se) and Mattias Olsson
Forsmarks Kraftgrupp AB, Chemistry and Radiochemistry, SE–742 03 Östhammar, Sweden

Alpha active nuclides, mostly actinides, can be released to the primary system in a nuclear power plant in case of degraded fuel failures. As the primary system is opened, contamination can be released to the air and a risk of internal contamination arises. Since alpha emitters have high relative biological effectiveness, they 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. Due to the long half-lives of most actinides, historical fuel failures could have a long term impact on the contamination levels. The very short range of alpha particles makes them difficult to detect and therefore it is difficult to determine the amount of activity inside a piping system.

To investigate the extent of the potential airborne alpha activity at Forsmark, air filters from mobile sampling boxes that monitor aerosols in the ventilation system were used. Three different parts of the ventilation system were included in the study: the ventilation from areas most likely to contain airborne activity before filtration and the unfiltered outgoing ventilation from the reactor building and turbine building respectively. Selected air filters from Forsmark units 1–3 were leached and analysed by alpha spectroscopy for nuclide specific determination.

The level of airborne alpha activity was low at all sampling points, approximately 10^{-5} Bq/m^3 in total. The highest activity, by a factor of ten, was found for the alpha energy 5.2 MeV which corresponds to Polonium-210 (Po) and Americium-243 decay. Due to the limitations of the sample preparation procedure and the average resolution of alpha spectroscopy it is not possible to distinguish between the two nuclides. However, the major contributor to the peak is most likely Po-210 which is a progeny of Radon-222. The second most prevailing peak was that of Plutonium-238 and Americium-241. Plutonium-239/240, Curium-242 and -244 could be detected at most sampling points, whereas Uranium-235, -236 and -238 were present at levels too low to detect.

Even though there are large differences in the tramp uranium history between the three units, this was not distinctly reflected in the detected levels of alpha activity in air. The relatively short-lived Curium-242 (163 days) was however found more frequently at the unit with the most recent fuel failure with accompanying dissolution of fuel. No clear differences in activity level or detected nuclides could be seen between the different parts of the ventilation system. The air filters were also analysed by gamma spectroscopy to determine the levels of gamma emitting nuclides, mainly Cobalt-60. No correlation could be seen between high levels of gamma emitting aerosols and the levels of alpha activity.

To conclude, the level of alpha airborne radioactivity at Forsmark NPP is low. The main contributor to the activity is caused by a progeny of natural occurring Radon. However, it should be noted that the oxide layers inside the primary system is still a potential source for internal contamination of alpha emitters.

POLARIS GAMMA-RAY IMAGING SPECTROMETERS

Zhong He

University of Michigan, Ann Arbor, Michigan 48109, USA
hezhong@umich.edu

Room-temperature 3-dimensional position-sensitive CdZnTe detectors, pioneered and developed by University of Michigan, can provide high spectroscopic resolution close to that of high purity germanium detectors. The elimination of cryogenic cooling makes CdZnTe detectors much more convenient for applications in nuclear facilities. In addition, 3-D CdZnTe detectors can image gamma-ray sources in the entire 4-π field of view without any blind spot in real time with high imaging efficiency. These systems enable the operator to see radiation dose distribution surrounding the detector or to see distributions of different isotopes simultaneously. Polaris systems at University of Michigan can detect and locate gamma-ray sources an order of magnitude weaker than natural gamma-ray background, making them ideal for detecting, locating and analyzing gamma-emitting sources in nuclear power plants, for environmental clean-up applications, and for radiation protection monitoring. This presentation introduces Polaris 3-D CdZnTe detector technology, and demonstrates how to obtain 3-dimensional distribution of isotopes in a room using Polaris CdZnTe detectors.

EPRI ALPHA MONITORING AND CONTROL GUIDELINES, REVISION 2 (EPRI report 3002000409)

Phung Tran, Senior Project Manager

EPRI (Electric Power Research Institute)
3420 Hillview Avenue
Palo Alto, CA 94304 U.S.A.
(650) 855-2158
ptran@epri.com

In 2006, the Electric Power Research Institute (EPRI) first published the *EPRI Alpha Monitoring Guidelines for Operating Nuclear Stations* (EPRI report 1013509), to provide standardized guidance for monitoring alpha contamination. Minor revisions were made to the guidelines, which were re-issued in 2009 (1019500). Most US and some international utilities have implemented the guidelines, and several areas of improvement were identified to enhance information to support the monitoring and protection of workers. EPRI has performed a major revision of the document to provide the necessary additional guidance to reflect more recent operating experience and to address program gaps.

This document provides guidance on how to identify the presence of alpha emitting radionuclides in operating nuclear reactors; a risk informed, graded approach to monitoring alpha emitting radionuclides based on the relative abundance of alpha emitters compared to the beta-gamma emitters; and guidelines on how to protect and train workers and how to monitor individuals for exposure to alpha emitting radionuclides. To support guideline implementation, the document has several appendices, including a summary of the monitoring guidelines, information on source term assessments, the technical bases of the guidelines, radon compensation, and instrumentation with examples for work control and internal dose assessment.

This presentation will discuss the revisions made to the guideline for enhanced alpha monitoring and control in operating nuclear power plant environments.

REGULATORY SUPERVISION OF RESPIRATORY PROTECTION

Petra Hansson
Swedish Radiation Safety Authority
SE-171 16 Stockholm
Solna strandväg 96
Tel: +46 8 799 44 38
Fax: +46 8 799 40 10
E-mail: petra.hansson@ssm.se

The Swedish Radiation Safety Authority works proactively and preventively in order to protect people and the environment from the undesirable effects of radiation, now and in the future.

Therefore regulatory supervision of respiratory protection is included as a natural part of the Swedish Radiation Safety Authority´s activities to ensure that the responsible party conducts the activity in a safe manner. Respiratory protection is one of many protective equipment regularly used in radiation protection activities. It is therefore important to understand how to use and maintain respiratory protection. It’s also important that the workers get proper training how to use and wear the respiratory protection in order to provide adequate protection against inhaling radioactive substances that can lead to internal contamination. The authority inspects, among other details;

- Respiratory protection management
- Priorities in the daily maintenance and decontamination of respiratory protection
- Information to the workers about respiratory protection

The presentation will cover the findings from the inspection and summarize the experience gained.

SPHERE CAGE: REINFORCEMENT OF THE MEANS OF DYNAMIC CONTAINMENT TO GUARANTEE THE HABITABILTY OF THE REACTOR BUILDING DURING OUTAGES IN EDF NUCLEAR POWER PLANTS

[1]MAURER Jean-Eric, [2]BIGNON Patrick, [2]FOY Gérald, [2]PIEGAY Eric

1 : EDF CNPE du Tricastin BP 9, 26130, St Paul 3 Châteaux, France
2 : S.A.S DELTANEU ZI rue Ampère, 59933, La chapelle d'Armentières, France

Some events of airborne contamination occurred in reactor buildings in the maintenance operations during outages in France and around the world. These events can have a health impact on workers and are treated by evacuation of the reactor building until the radiological conditions are back to normal. These evacuations have a direct impact on the planning and organisation of resources. The source term of airborne contamination is generally associated with the loss of a dynamic containment in a work field. The risk for activities can be: eddy current testing on the steam generators, maintenance of a check valve or a large section tap, cutting of a pipe portion or a high-pressure cleaning operation of a highly contaminated capacity.

The analysis of the origin of the main events encountered on the fleet of 58 French reactors shows that 90% of them are related to a failure of exploitation of containment devices dynamics on the fields. These defects include brutal breakdowns of Portable Ventilation Units by power shut off or damages of the filter medias. If the design of the new Portable Ventilation Units "advanced / second generation" has answered to these defaults, some sources of default still remained as the obstruction of the Portable Ventilation Unit's inlets. The obstruction of the suction duct can cause a depression and its sudden dash pot which can cause a ripping of the containment tent.

The observation on the field has highlighted hazardous situations like: first of all the dispersing risk of contamination. Then the risk of a fire due to overheating of PVU and finally a security risk for workers wearing airproof ventilated clothes.
To reduce the occurrence of these events, TRICASTIN NPP has developed a mean of securing the suction PVU used in EDF: The sphere cage. After a modeling study of ventilation phenomena validated by the engineering office of an industrial company, the sphere cage has been developed. This model could be adapted in accordance with suction ducts of different diameters and different flows. This design requires no power source and is a very fast and instinctive facility for logisticians. It is a pragmatic, simple and effective answer to these events.

The implementation of this system reinforces increasingly the dynamic containment. It has been used on fields with high risks and has been successful in eradicating these situations. Also the workers are more serene during their interventions.

With this generalisation, the expected gains are safety, security and health of workforce, the duration of the outage and resource management.

HOT-SPOT MANAGEMENT AT NPP LEIBSTADT

Lars Kämpfer
Radiation Protection Manager, NPP Leibstadt, Switzerland
lars.kaempfer@kkl.ch

In 2009 after an overexposure of persons due to unexpected high dose rates in a Swiss NPP the person in charge for radiation protection in NPP Leibstadt started an overall check about the provisions against high dose expositions. As a result several improvements were realized. This presentation will show and discuss the main aspects on design, processes and procedures how NPP Leibstadt protect persons from exposition by potential high dose rate areas and hot spots.

RADIATION PROTECTION ASPECTS OF FUEL HANDLING INCIDENT AT FORSMARK NPP

Staffan Hennigor, SIG@forsmark.vattenfall.se; Björn Brunefors BBS@forsmark.vattenfall.se:
Patrick Brandelind BRP@forsmark.vattenfall.se: Frida Svensson SVF@forsmark.vattenfall.se
Forsmarks Kraftgrupp AB, SE-742 03 Östhammar, Sweden

A damaged fuel element was in the position to remove the fuel bundle from the fuel box.
The fuel element was not to be reused but some fuel rods should be sent for material test.

The fuel bundle was however stuck after being pulled approximately 1 meter outside the box.
When trying to lower the fuel back into the fuel box it again got jammed. One of the spacers was damage during this maneuver.

A risk assessment was performed and to prevent the fuel from falling it was secured with a screw through the box. Securing and collecting debris from spacer and box in addition to calculation of possible release of iodide and noble gases were some of the radiation protection challenges during this incident.

In parallel; a graded approach with respect to time restrictions for the handling of used fuel has been developed, depending on what kind of work will be performed.

SITE-SPECIFIC RADIONUCLIDE DISTRIBUTION DURING REFUELLING OUTAGE

B. BREZNIK, L. MIKELIĆ, M. PAVLIN, M. URBANČ
Krško NPP
Vrbina 12, 8270 Krško – Slovenia

In October 2013 the refuelling outage started in the Krško NPP. Primary chemistry results during operation indicated two open fuel cladding defects. The first smear from an open valve showed 0.4% of gross alpha contamination. The first measurements of dose rates at primary system loops were also higher than before. After these indications, RP Superintendent issued a leaflet with the information and instructions on protection measures to avoid alpha risk during outage activities. Plant radiation safety committee approved these instructions and provided further information on work and ALARA planning at the beginning of the outage. Later on visual inspection of all fuel assemblies showed nine open defects/broken fuel rods of four fuel assemblies attributed to baffle jetting at the core baffle locations.

Health physics team collected smear samples of loose surface contamination from plant components of the primary systems and of some exposed areas in the containment. A few highly radioactive crud spots were collected and analysed. The poster will present source term assessment and site-specific radionuclide distribution during refuelling outage. The analysis of the smears and hot spots has helped to confirm a mechanism of fuel cladding failures and reasons for higher radiation levels found at some locations.

Alpha contamination of the air was being monitored continuously in the containment and kept under control also by individual air samplers when necessary. Alpha radionuclide distribution was established based on samples collected by a few smears. This information has been prepared to evaluate possible alpha intake in case of positive results of internal gamma contamination. All internal doses were evaluated to be below detection or registration limit.

STUDY OF THE ^{133}XE EXPOSURE

Aurélien PERIER[1,2], Estelle COURAGEOT[1], Emmanuelle GAILLARD-LECANU[1], Gaëlle LE-MEUR[1], Catherine MONIER[1], Dominique THERS[2]

[1] EDF R&D, 6 quai Watier, 78401 Chatou, France
[2] Subatech, Ecole des Mines, CNRS/IN2P3 and University of Nantes, France

During the nuclear reactor operation, fission products and radioactive inert gases, as ^{133}Xe, are generated and might be responsible for the exposure of workers. For an exposure to radioactive noble gases, the external exposure is the only exposure mode taken into account to quantify the dose to the whole-body, received by workers (ICRP, 1978).
Studies on the external exposure have been done for an immersion in an infinite cloud of ^{133}Xe but not in a realistic environment such as the reactor building of a nuclear power plant (Poston and Snyder, 1974; Piltingsrud and Gels, 1985; Eckerman and Ryman, 1993; ICRP, 1994).

The study has been conducted to update data on the exposure to ^{133}Xe by using Monte-Carlo simulations based on GEANT4 (Allison and al, 2006), the MIRD phantom (Cristy and Eckerman, 1987) and a realistic geometry of the reactor building.
In these simulations, the phantom MIRD has been adapted to the problem of the external exposure in a ^{133}Xe cloud. Organs have been added inside the phantom, such as esophagus, salivary glands, eyes, lens and its radiosensitive part (ICRP, 2002; ICRP, 2009).
GEANT4 is used to simulate the radiation interaction with matter. The ionizing radiation resulting from disintegrations of ^{133}Xe is tracked in the program to get the energy deposed in all the organs of the phantom. The equivalent dose rate in the organs and the effective dose rate are determined by using the weighting factors (ICRP, 1991).

In this study, the aim is to obtain a dose assessment in a whole reactor building containing ^{133}Xe, assuming that the worker stands up on the pool floor of the reactor building.
Firstly, as several exposure scenarios are possible on the pool floor, the environment of the worker is studied to find the place where the maximum dose rate is in the reactor building.
Secondly, the phantom stands up in this place to treat the ^{133}Xe external exposure in the whole reactor building. However, the air volume of the whole reactor building is huge to be simulated once. Consequently the ^{133}Xe source contained in the reactor building is divided in several hemispherical shells of 2-meter-thickness in which the disintegrations of ^{133}Xe are confined. This simulation technique allows following the exposure in the reactor building coming from a hemispherical shell of ^{133}Xe. A calculation is achieved to get the effective dose rate in the whole reactor building.
Then, another study is completed to evaluate the effective dose rate for an immersion in a semi-infinite cloud of ^{133}Xe. This value is compared with the reference data in order to check the consistency of our result.
Furthermore, due to the particular emission spectrum of ^{133}Xe, the exposure to eye lens in the reactor building is determined by Monte-Carlo simulations. The equivalent dose rate to the lens of eyes is discussed in the light of expected new eye dose limits.

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ONGOING WORK TO ENHANCE POST-ACCIDENT RADIATION PROTECTION AT SWEDISH NUCLEAR POWER PLANTS

K. Fritioff[1], S. Hennigor[2], M. Carlsson[3], C. Solstrand[4], I. Svantesson[5]

[1]Vattenfall AB, karin.fritioff@vattenfall.se
[2]Forsmarks Kraftgrupp AB
[3]Ringhals AB
[4]OKG Aktiebolag,
[5]SKB AB

The Nuclear power plants in Sweden have established a working group with the objective to identify areas for improvement and develop action plans for the radiation protection capabilities during severe accidents following a station black out. An overarching goal of the group is to harmonize, when needed and when possible, the strategies and procedures within post-accident radiation protection between the nuclear facilities. A common approach will facilitate the assistance from the others and the communication and interaction with relevant authorities.

The working group have identified the following key areas:

- Personnel (staffing of EP organization) & access to equipment
    - Contracts with external suppliers – RP technicians & equipment
    - Cooperation between Swedish NPPs – personnel & equipment
    - Mobile equipment stored close to the site area
- Training & Exercises
    - More practical and theoretical exercises
    - Monitoring exercises
    - Cooperation between Swedish NPPs
- Strategies and robust procedures for post-accident radiation protection
    - Doscriteria (Dose management)
    - Operational intervention levels
    - Personal dosimetry (External / internal)
    - Security procedures have to be changed
    - Evacuation of the NPP
- Availability of areas within the NPP
    - Strategies for on-site work
    - Updating of post-accident layouts
    - More radiation monitors in the reactor buildings
- Logistic center
    - Receive equipment & personnel and supply protective equipment
    - Screening for external and internal contamination, cleaning
    - Mobile measuring lab

The presentation will discuss the most prioritized areas and activities of the above; as it is today and as it will be tomorrow.

ISOE EXPERT GROUP ON OCCUPATIONAL RADIATION PROTECTION IN SEVERE ACCIDENT MANAGEMENT

Halil Burçin Okyar, Ted Lazo
OECD Nuclear Energy Agency
Issy-les-Moulineaux, France
Tel: +33 1 45 24 10 45
Email: halilburcin.okyar@oecd.org, edward.lazo@oecd.org

Since its creation in 1992, the Information System on Occupational Exposure (ISOE), jointly sponsored by the IAEA, has facilitated the exchange of data, analysis, lessons and experience in occupational radiation protection at nuclear power plants worldwide. It maintains the world's largest occupational exposure database and a network of utility and regulatory authority. ISOE is a Joint Project under the NEA's statute, and does not report directly to Committee on Radiation protection and Public Health (CRPPH) nor request its approval for its programme of work; however the ISOE programme cooperates extensively with the CRPPH on the Committee's operational radiological protection work.

The Expert Group on Occupational Radiation Protection in Severe Accident Management and Post-Accident Recovery (EG-SAM) has been established to develop a report on best radiation protection management procedures for proper radiation protection job coverage during severe accident initial response and recovery efforts to identify good radiation protection practices and to organise and communicate radiation protection lessons learned from previous reactor accidents. An interim report with a general perspective and discussion of specific severe accident management worker dose issues will be finalized until the end of 2013 and the report will be finalized by organizing an international workshop in 2014. This interim report focuses on radiation protection management and organization, training and exercises related to severe accident management, facility configuration and readiness, worker protection, radioactive materials, contamination controls and logistics and key lessons learned from Chernobyl and Fukushima Dai-ichi accidents.

The presentation will introduce the details of the report and summarize the NEA contribution to work of the expert group.

ONLINE MULTICHANNEL DOSIMETRY DEDICATED TO RADIATION PROTECTION AND DECOMMISSIONING OF NUCLEAR FACILITIES

Sylvain Magne, Karim Boudergui
CEA, LIST, Laboratoire Capteurs Archite
91191 Gif-sur-Yvette, France
Tel: +33 (0)1 69 08 96 86 Fax: +33 (0)1 69 08 60 30 karim.boudergui@cea.fr

Optically Stimulated Luminescence (OSL) is investigated at CEA LIST since 1995 for online remote gamma/beta dosimetry. It was successfully applied to decommissioning of nuclear installations (*e.g.* CEA-Atelier Pilote de Marcoule and AREVA-UP1 plutonium production center (both in Marcoule, France)) and more recently to *in vivo* dosimetry during radiation therapy treatments [1].

OSL dosimetry relies on storage materials (*i.e.* passive dosimetry) that are sensitive to light. OSL sensors are connected to optical fiber links and monitored online. After ambient exposition for a given amount of time, the electrons stored within the OSL material are released under light stimulation and the subsequent OSL is collected and guided back along the same fiber up to the monitoring unit placed away (in safe zone). OSL signals are then corrected for background noise and integrated to provide the absorbed doses. After stimulation, the sensor is reset and re-used again for another integration period. As an example, a dose resolution of 20 µGy has been achieved with an OSL alumina crystal connected to a 20-meter long fibre [2].

In practice, the dose range lies typically between 3 and 3.5 orders of magnitude (*e.g.* between ~ 0.2 mGy and several Gy). The operator also has to choose the integration time according to expected activity, between 2 and 2.5 orders of magnitude (from 10 minutes up to several days). For "high-activity" monitoring (> 1 Gy/h), the integration time should not be too high to avoid dose saturation (*e.g.* < 1 hour). Conversely, for "low-activity" monitoring (< 100 µGy/h), a low resolution in dose rate may be achieved at the expense of long integration times.

By combining both dose/time parameters, the range in dose rate may be as high as 6 orders of magnitude. For decommissioning applications for instance, OSL sensors are usually left in place overnight and read after a 16-hour integration period, thus leading to a resolution in dose rate of typically 200 µGy/16 h, *i.e* 12.5 µGy/h. Therefore, OSL sensors provide an interesting alternative to traditional Geiger-Muller detectors that are usually more bulky and rigid, require high voltage to operate and are not prone to multiplexing into a single cable.

At CEA LIST, both Alkaline-Earth Sulfides (AES) polycrystals doped with Eu-Sm ions provided by Université de Montpellier, France and α-Al_2O_3:C crystals provided by Landauer, USA were used. The CEA LIST has developed a multichannel OSL Fibre Optic (OSL/FO) reader for Radiation therapy purposes [3] that may be used as well for decommissioning and radiation protection in nuclear facilities. Such instrument was built in 6U-19 inch enclosure, handled by a laptop (USB). It includes a laser, a 16-channel optical switch and a photomultiplier stage operating in photon-counting mode. The OSL fibre sensors are linked to flexible fibre cables and incorporate millimetric-size Al_2O_3:C crystals (high sensitivity and linearity, no fading at room temperature, low Z (~ 11)). They are radiation-resistant, electromagnetic-immune and exhibit low energy and angular dependencies [4]. The readout time is about 30 seconds for each sensor (8 minutes for 16 sensors).

Finally, a compensation technique for alumina crystals was designed and patented for radiation protection applications in order to meet IEC 61066 standards in terms of tissue-equivalence (energy range [15 keV, 3 MeV] and angular incidence = ± 60°) using a suitable combination of metallic filters. Taking advantage of circular symmetry (with respect to fiber axis), an Al_2O_3-OSL fiber sensor may perform over a wider range of $\pm$ 120° (i.e. 95 % of space (4π Sr)).

In conclusion, the CEA LIST has developed a multichannel OSL unit and fiber sensors that may be assembled in cable to provide multipoint sensing (1-D cartography) inside hard-to-access locations (*e.g.* pipes, tanks). In the electronuclear industry, targeted applications are radiation protection at fixed points into nuclear facilities and decommissioning of nuclear installations.

This work has been partially funded by AREVA-NC, the European Commission (MAESTRO Project) and the French Agence Nationale pour la Recherche (ANR CODOFER and INTRADOSE).

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A LARGE VOLUME AND ROOM TEMPERATURE GAMMA SPECTROMETER FOR ENVIRONMENTAL RADIATION MONITORING

R. Coulon, T. Thit, K. Boudergui, S. Normand

CEA, LIST, Laboratoire Capteurs et Architectures Électroniques, 91191 Gif-sur-Yvette Cedex, France

Room temperature gamma spectrometer is an issue for environmental radiation monitoring. Indeed to monitor nuclear isotopes releases around a nuclear power plant convenient instruments giving fast and useful information are required.

Gamma spectroscopy gives isotopic information about radioactivity. This identification allows then to make a diagnostic on the situation and notably on the nature of the radioactive materials. To separate isotopes with high efficiency and reliability the energy resolution has to be as low as possible. The lowest resolution is achieved by cryogenic detectors (HPGe) which are not appropriate for outdoor condition of use. Room temperature gamma spectrometer with quite good resolution has been developed last decade. The CdZnTe semiconductors and LaBr3:Ce scintillators provide an energy resolution about 2-3 % at 662 keV. However they are limited in size typically 1.5 cm^3 for CdZnTe and 232 cm^3 for LaBr3:Ce.

High pressure xenon chamber HPXe could achieve an energy resolution about 2-3 % at 662 keV and a density of 0.6 $g.cm^{-3}$. Detection performances are then close to CdZnTe and LaBr3:Ce technologies in terms of energy resolution and radiation stopping power. The advantage of gas detectors compared to other detector technologies is the possibility to build large volume sensor (about several liters) and that it is rad-hard by the use of gas. Such a large volume detector will show an improvement in detection efficiency and peak to Compton ratio. Therefore detection and identification could be then improved by an order of magnitude compared to other room temperature spectrometers for environmental radiation monitoring.

To achieve an energy resolution below 3 %, the gas has to be highly purified (<1 ppb impurities) but the remaining difficulty for large volume HPXe chambers is the ballistic effect due to ions motion impact on electrons collection. Methods allowing the management of this ballistic effect have been developed in numbers of research teams (Firsch grid, PMT coupling, virtual grid…). This paper presents the stat of the art about HPXe technologies and an approach to develop large area room temperature spectrometers.

IMPACT OF MEASUREMENT LOCATION ON THE HOT LEG AND COLD LEG DOSE RATES

Benfarah M.[a], Picaud B.[a], D'Ascenzo L.[b], Dacquait F.[c]

[a:] *EDF, SEPTEN, 12-14 av Dutrievoz, Villeurbanne, F-69628 France.*
Moez.benfarah@edf.fr +33 4 72827059
[b:] *CEPN, 28 rue de la Redoute, Fontenay-Aux-Roses, F-92260, France.*
[c:] *CEA, DEN, Cadarache, Saint-Paul-Lez-Durance, F-13108, France*

Several utilities use indicators or contamination monitoring programs based in dose rates measured on the primary circuits of pressurised water reactors. The study analyses the impact of the location of the measurement points on the dose rate values of the hot legs and the cold legs of the four loops and three loops PWR. It is based on the feedback of the measurements carried out on the EDF fleet as well as on theoretical calculations using structure activation code and radiation propagation code.

The feedback has shown that the hot leg dose rate is higher when the measuring point is closer to the reactor. The hot leg dose rate is lower when the measurement point is located on the side of the steam generator. The gamma spectrometry measurements have shown that there is a gradient of 60Co activity all along the hot leg resulting from the neutron activation of the structure of the hot leg. The impact of the neutron activation on the dose rates of the primary circuit has also been demonstrated by the theoretical calculation.

Therefore, for a relevant comparison of hot and cold leg dose rates measured at different PWR units, the location of the measurement points must be precisely identified and indicated. The comparison is relevant only when the measurement locations are comparable.

SIMULATIONS AND EXPERIMENTATIONS TO QUANTIFY PARTICLES RESUSPENSION RATE DUE TO HUMAN WALKING IN RADIOACTIVE CONTAMINATED AREA

Z. Mana[1,2,3], F. Gensdarmes[1], L. Ricciardi[1], M. Rabaud[2], S. Jahan[3], G. LeMeur[3], C. Monier[3], F. Drouet[4], H. Onillon[3], F. Feuillebois[5]

[1]Institut de Radioprotection et de Sûreté Nucléaire (IRSN), Gif-sur-Yvette, 91192, France
[2]Fluides, Automatique et Systèmes Thermiques, University of Paris-Sud, Orsay, 91405, France
[3]Electricité de France, EDF R&D, Chatou, 78401, France
[4]Electricité de France, EDF DPN UNIE, Saint Denis, 93282, France
[5]Laboratoire d'Informatique pour la Mécanique et les Sciences de l'Ingénieur, University of Paris-Sud, Orsay, 91405, France

Introduction

In nuclear facilities, during normal operations in controlled areas, workers could be exposed to radioactive aerosols. One of the airborne contamination sources are particles initially seeded on the floor and that could be removed by workers while they are walking. In the future, EDF will increase its maintenance operations (Grand Carénage Project) which will involve co-activity of the workers. In order to assess occupational exposure and protect the workers, it is suitable to determine accurately particle resuspension rate. Up to now, the estimation of particle resuspension rates is based on empirical considerations. Therefore, this study will adapt a resuspension model to human walking case in order to estimate particle concentration in facilities.

Methodology

To estimate particle resuspension, Rock'n'Roll model is selected since it describes very well aerodynamic resuspension mechanisms and integrate both drag and lift aerodynamic forces. This model requires two parameters: the airflow forces distribution and the adhesion forces distribution.

To estimate these two parameters, on one hand, CFD (Computational Fluid Dynamics) simulations are performed to estimate airflow and friction velocity under the shoes. On the other hand, AFM (Atomic Force Microscopy) measurements allow to access accurately to adhesion forces distribution. Model results are then compared to analytical experiments performed on a controlled area. A mass balance equation gives particle resuspension rates from particle concentration measured in this controlled environment. We are studying cobalt oxide particles resuspension on an epoxy surface and the results are compared to a reference case: alumina particles on a glass surface.

Results

To fit to the human walking case, the walking motion was studied to estimate shoes rotation speed. This speed is an input parameter for CFD simulations. First simulations were performed for a simple geometry: a rotating plate of 30 cm large. For an angular speed of 200 °.s-1, the airflow velocity under the plate reached 12 m.s-1. The results are compared to PIV (Particle Image Velocimetry) measurements for the same plate and the agreement is satisfactory. Then CFD simulations involving safety shoes are underway. The airflow under the shoes is investigated by numerical simulations in order to access to airflow speed and friction velocity.

The second parameter depends on roughness and particle diameter. First, the size of the particle was examined based on samples coming from an EDF nuclear facility. Particles between 1 µm and 10 µm are present in a large proportion. Moreover, AFM (Atomic Force Microscopy) measurements are underway to estimate particle adhesion distribution and its evolution for different values of surface roughness.

On the same time, experiments that simulate human cycle walking are done to highlight the influence of surface roughness, relative humidity, walking velocity and different surface particle concentrations on particle resuspension rate. The results of these experiments will be compared to the Rock’n’Roll model’s ones.

Reeks, M.W. and Hall, D. (2001) Kinetic models for particle resuspension in turbulent flows: theory and measurement. Journal of Aerosol Science, 32, 1-31

GERMAN NUCLEAR POWER PLANTS TRENDS IN OCCUPATIONAL EXPOSURE

Thorsten STAHL, Claudia SCHMIDT
Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS) mbH, Germany

After the Fukushima accident, eight nuclear power plants in Germany were finally shut down due to the amendment of the Atomic Energy Act of July 2011. Also the remaining nine plants will be finally shut down in a stepwise process until 2022. In addition, 16 nuclear power plants are under decommissioning.

The presentation will provide an overview of the occupational exposure in the German nuclear power plants in operation, in the transition phase and under decommissioning. Current trends and in particular differences between the exposure situations for these stages of the nuclear power plants will be discussed.

FULL SYSTEM DECONTAMINATION
- The key approach to Decommissioning -

Topf, Dr. Christian, Sempere-Belda, Luis, Basu, Ashim
AREVA NP GmbH, Paul-Gossen-Str. 100 / 91052 Erlangen, Germany,
christian.topf@areva.com, luis.sempere-belda@areva.com, ashim.basu@areva.com

Worldwide more and more nuclear power plants are reaching the end of their operational life, thus entering the post-operational phase prior to decommissioning.

In Germany, the implementation of the 13th amendment to the German Atomic Energy Act (Atomgesetz, AtG) as a direct consequence of the events in Fukushima led to a complete change in the German national nuclear policy. The decision to permanently shut down eight NPPs with immediate effect while phasing out the remaining NPPs by 2022 at the latest, poses great challenges to German utilities in terms of planning and implementing economically reasonable decommissioning and dismantling (D&D) concepts after the post-operation phase.

As a preparatory measure prior to decommissioning, a Decontamination applied during the post-operation phase is the most accepted approach worldwide. Following the ALARA principle, the implementation of a Full System Decontamination (FSD) - defined as the chemical decontamination of the primary system incl. RPV in conjunction with the main auxiliary systems - provides a variety of advantages not only for future decommissioning but for the post operational phase in particular:

- Minimization of radioactive inventory at an early stage
- Minimization of personnel dose exposure during planning (post-operational phase) and performance of D&D
- Minimization of radiological sampling for dismantling, planning for rad-waste and storage
- Minimization of nuclide vector numbers
- Increased flexibility in decommissioning strategy
- Support in dismantling techniques (e.g. for RPV)
- Reduction of waste for post treatment and increased free release rate or very low level waste volumes
- Reduction of long-term cost for rad-waste treatment and storage
- Facilitation of the licensing process, thus, higher reliability of scheduling and budgeting

With its proprietary decontamination technology CORD® and AMDA® AREVA is most experienced in decontamination for decommissioning worldwide. This applies especially to Full System Decontamination (FSD). Since 1986, AREVA has performed more than 15 D&D Decontamination Projects after shutdown, immediately or years later and after reopening Safe Enclosure. AREVA is capable of assisting the German utilities in the planning and performance of DCDs for the current and future shutdown schedule.

This paper will outline AREVA's comprehensive approach in decontamination for decommissioning and highlights the experiences and results of the first two FSDs performed in the aftermath of the final shutdown decision at NPP Unterweser (KKU) and Neckarwestheim 1 (GKN 1) in Germany.

MODELLING THE RELATION BETWEEN THE ACTIVITY OF A PWR PRIMARY CIRCUIT AND THE OCCUPATIONAL DOSE

A. Artmann, S. Schneider[1)]
[1)] Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Germany

The occupational dose received by workers in a NPP with pressurized water reactor is determined by the activation and contamination of structural elements and parts of the primary circuit and the coolant. To limit radiation dose rates from activated components, actions can be taken only by using shielding. Another approach is to effectively reduce the contamination with chemical methods, leading to the potential to reduce occupational dose rates during revision outages and during decommissioning.

Data on occupational exposure in German NPPs as well as information about the radionuclide concentration in the coolant are available. Although there is an obvious link between activity and dose rates, credible quantitative predictions in complex systems like nuclear power plants are challenging. The complexity arises from the number of parameters involved, such as shield geometry, self-shielding of components, deposits of radionuclides, or the behaviour of working personnel.

The goal of our project is to fill the gap between activity and dose by using a generic and adaptable model of a PWR's primary circuit and its basic shield components. A 3-dimensional CAD-Model has been created to determine the geometrical input (distances and angles) for calculations within the MicroShield software. Combining the 3D-Model and the calculating tool, dose rates can be determined at definable positions in the surroundings of the primary circuit. By inserting the time variable, one can map specific jobs to combinations of stay times at the pre-defined positions. The accuracy of this generic and simplified model can be verified and optimised by comparison of calculations with actual data on job-related dose rates.

The simplified model can contribute to the reduction of radiation exposure for workers by showing up the potential of an optimisation of chemical decontamination in connection with revision outages and decommissioning actions.

IMPLEMENTATION OF THE PRINCIPLE OF OPTIMIZATION IN NUCLEAR INDUSTRY, DOSE CONSTRAINT, ALARA AND WORK MANAGEMENT

F. Mancini

Sogin Caorso NPP – via Enrico Fermi, 5/a – 29012 Caorso (Piacenza) – Italy
Tel. +39 3316576193 - Fax. +39 0683040477

All activities justified, involving the use of ionizing radiation, must respect the principle of optimization, namely the reduction of exposures at levels reasonably achievable taking into account economic and social factors. In application of this principle the ICRP since 1991 recommended the use of dose constraint concept strongly advocated in ICRP 103, which is a restriction on the individual dose referred to the "source" that serves the optimization of protection and represents the upper bound of expected dose for that source.
The selection of a numerical value of dose constraint for a source is not a simple task, it requires knowledge and experience and a thorough examination of the conditions of exposure.
The use of dose constraint does not allow itself to comply with the principle of optimization.
In the international arena, it is increasingly important to the concept of "work management", that is a methodology that highlights the importance of fully manage the work, from planning to follow-up, using a multi-disciplinary team approach that involves all parties concerned.
The application of work management involves careful planning and management of activities and the application of a well-defined ALARA program. This program reflects the commitment to properly implement radiation protection measures, define the objectives and describe the specific structures, procedures and tools necessary for their implementation.
Among the tools used the ICRP also indicates the cost-benefit analysis in which the benefits or effectiveness of an "option radioprotection" are quantified in monetary terms, associating a reference value of the monetary unit dose avoided: called "alpha value ".
This paper provides information on the evolution of the concept of dose constraint at international level, its current use for the optimization of radiation protection within the Member States of the European Union and similar information for Canada and the United States, a description of the key concepts in the work management and finally an international framework on the application of the cost-benefits.

IMPROVEMENTS IN TRANSURANIC NUCLIDES CONTAMINATION RISK MANAGEMENT AT CERNAVODA NPP

Catalina Chitu, Catalina.Chitu@cne.ro; Ion Popescu Ion.Popescu@cne.ro

"CNE Cernavoda" NPP, No. 2, Medgidiei Str. Cernavoda 905200, Romania

Some unexpected events in nuclear industry revealed several deficiencies in transuranic nuclides – TRU monitoring. Based on industry operating experience Cernavoda NPP Radiation Protection program was improved to identify and mitigate the risk of internal contamination with TRU. Internal dose assessment for intakes of TRUs is a difficult radiation protection task. Contamination monitors and whole body counting equipment used by Cernavoda NPP Dosimetry Laboratory to monitor personnel internal contamination are very effective at identifying individual exposed to gamma emitting radionuclides.
Every contamination event is investigated and internal contaminations with gamma emitters are promptly identified. An estimate of the intake of gamma emitting radioactivity is determined from these whole body count measurements and applicable intake retention fractions.
Main alpha contamination sources in radiological area were identified and representative samples analyzed for alpha nuclides qualitative and quantitative analysis in a specialized laboratory. Internal contamination from alpha emitters can be determined using scaling factors based on a representative radioactivity samples from the working area.
The present paper describes the main operational aspects of the management of TRU internal contaminations at CNE Cernavoda.

RADIATION MONITORING MEASURES AFTER THE FUKUSHIMA ACCIDENT IN BOHUNICE NPP

Ľubomír Dobiš

Slovenské elektrárne, a. s., subsidiary of Enel
Bohunice Nuclear Power Plant
91931 Jaslovské Bohunice
Slovak Republic

Tel.: +421 33 597 2128
Fax: +421 33 597 4761
E-mail: lubomir.dobis@enel.com

After the Fukushima accident nuclear power plants passed the stress tests. The results of the tests in Bohunice NPP were analyzed and necessary modifications in nuclear and radiation safety were approved.

Among the proposed modifications there were also the radiation monitoring measures. The suggested updates were in the field of the addition of missing measuring points, of the seismic reinforcement of existing radiation monitoring points in the NPP surroundings, of the improvement of monitoring data transmission as well as in the field of optimization of mutual support between the Slovakian NPPs.

The poster provides information about the accepted radiation measures and the status in their implementation.

POST-FUKUSHIMA RADIATION MONITORING UPGRADE

B. BREZNIK, S. FUKS, M. PAVLIN, D. VEHOVAR
Krško NPP
Vrbina 12, 8270 Krško – Slovenia

An assessment of the possibilities for improving the Krško NPP on-site monitoring network after the Fukushima accident resulted in the specification for design and purchasing a new autonomous system of radiation monitors (ARMS) in year 2011. The system was designed for normal day-to-day use and also for extreme accidental situations. It is in operation since beginning of 2012.

The ARMS is composed of eight self-powered perimeter dose rate probes, three beta/alpha aerosol monitors in the yard, and additional system of self-powered dose rate probes inside the rooms of the radiological controlled area (RCA). The monitors in the RCA are installed with local alarm at eight locations in the auxiliary building which could be visited by the operators during post-accident conditions. Dose rate monitors are resistant to harsh environmental and seismic conditions and cover the range from 10 nSv/h to 10 Sv/h. Aerosol monitors are of standard design and range from 10^{-2} to 10^{5} Bq/m^3 beta.

These monitors are connected into wireless short-range network for data transmission to a dedicated central computer in the health physics office and also to plant process information system (PIS) which is used by plant operators. The signal from perimeter dose rate and airborne monitors can be easily monitored by the receiver installed in an emergency monitoring van, up to a few kilometres from the station. This mobile unit has also a spare dose rate monitor as a possibility for a new off-site location in case of an emergency.

Dose rate detectors have internal power supply with battery capacity of five to ten years during normal conditions; receivers installed in the RCA and the central computer unit have a reliable power source.

With the use of specific software applications provided by the ARMS supplier, data from the MS SQL database can be presented to users in different ways such as current values, alarm values, event log and trends. For the measurements done with the probe inside the monitoring van, it is possible to follow its geographical location. For the off-site dose control, the information from ARMS and already existing off-site monitoring stations has been integrated within a common PIS screen.

Radiation protection technicians have also a possibility to use a set of additional handheld gamma radiation probes which communicate wirelessly inside the RCA or outside the buildings. During the next outage there will be a local network available for these small detectors of the measurement range from 0.5/10 µSv/h to 10/100 mSv/h inside the reactor containment.

RADIATION PROTECTION ASPECTS OF FUEL HANDLING INCIDENT AT FORSMARK NPP

Staffan Hennigor, SIG@forsmark.vattenfall.se; Björn Brunefors BBS@forsmark.vattenfall.se:
Patrick Brandelind BRP@forsmark.vattenfall.se: Frida Svensson SVF@forsmark.vattenfall.se
Forsmarks Kraftgrupp AB, SE-742 03 Östhammar, Sweden

A damaged fuel element was in the position to remove the fuel bundle from the fuel box.
The fuel element was not to be reused but some fuel rods should be sent for material test.

The fuel bundle was however stuck after being pulled approximately 1 meter outside the box.
When trying to lower the fuel back into the fuel box it again got jammed. One of the spacers was damage during this maneuver.

A risk assessment was performed and to prevent the fuel from falling it was secured with a screw through the box. Securing and collecting debris from spacer and box in addition to calculation of possible release of iodide and noble gases were some of the radiation protection challenges during this incident.

In parallel; a graded approach with respect to time restrictions for the handling of used fuel has been developed, depending on what kind of work will be performed.

DOSE TO THE LENS OF THE EYE - PRACTICAL INVESTIGATIONS FORSMARK NPP

Nastaran Farahi, Anna Svensson

Forsmarks Kraftgrupp AB
742 03 Östhammar, Sweden
Phone: +46 (0)706 78 1925

Nastaran.farahi@forsmark.vattenfall.com

The International Commission on Radiological Protection (ICRP) recently lowered their recommended occupational eye lens dose limit from 150 mSv in a year to 20 mSv in a year, averaged over a defined period of 5 years.

The project, "*Dose to the lens of the eye at Forsmark NPP*" is part of a larger project with participation from all Swedish NPP's. An abstract on this overall project has also been sent by Lisa Bäckström (**Vattenfall Research & Development AB**'s).

The measuring of the dose to the lens of 28 workers among the workers who has a high risk to receive higher eye lens dose in different job categories has been per formed. They have worn headband dosimeter for measurements of Hp(3). In addition, Forsmark's TL-dosimeters have a tape on the opening window for beta radiation so we will compare the beta dose result from personal dosimeter and Eye dosimeter. We will discuss how dosimeters are used in practical work situations and as well as the bias and difficulty that occurs in this study.

ALARA IMPLAMANTATION PRINCIPLES AT THE DESIGN, OPERATION AND DECOMMISSIONING STAGES AT ANPP

Aida Avetisyan
Armenian Nuclear Regulatory Authority
a.avetisyan@anra.am

The Armenian Nuclear Power Plant (ANPP), the only nuclear power plant in the region, consists of two VVER/440/270 units (that is a modified, seismic design VVER/440/230). Unit 1 started its commercial operation in 1976 and Unit 2 in 1980. Both units were shouted down shortly after the 1988 Spitak earthquake. Re-commissioning works were performed from 1993 to 1995 and in November 1995 the Unit 2 restarted operation. At this moment the ANPP Unit N1 is in conservation regime (long-term shut down). The new unit construction of ANPP is approved by Government of Armenia and the safety case preparation stage has been started.

The Armenian regulations on nuclear and radiation safety clearly defined the ALARA implementation at the design, operation and decommissioning stages.

Radiation Protection at the Design Stage of Nuclear Power Plants

The regulations stated that during designing of the protective shields against external exposure to personnel and population the safety factor equal to 2 should be accepted when estimating the annual effective dose. It is necessary also to take into consideration the availability of other radiation sources and further increase of their capacity. The design of protective shields against external exposure shall take into account premises, category of exposed persons and duration of exposure.

For calculation of the authorized discharges and releases from atomic energy utilization installations intended for practices with radiation sources, including nuclear facilities, it is accepted that the effective dose from annual releases and discharges during lifetime (70 years) does not exceed the dose constrain value. To ensure radiation protection, NPP design shall identify all real and potential sources of ionizing radiation and shall provide measures for ensuring the necessary technical and administrative control over their use.

The requirements with regard to the classification of zones and compartments, radiation monitoring, the individual protection means and the access control are established by a different regulation.

To keep the exposure of personnel and public as low as reasonably achievable during plant operation, the design of the reactor coolant system shall arrange for:

1. use of structural materials with minimum content of chemical elements with high activation cross-section and producing long-living radioactive corrosion products;
2. coolant purification from fission and corrosion products;
3. water chemistry control;
4. minimum length of the pipelines with a minimum number of isolation valves and connections;
5. leak-tightness testing of operating components;
6. decontamination of SSCs outer and inner surfaces;
7. prevention of uncontrolled radioactive leaks in the NPP premises.

The layout of the plant, its buildings and SSCs shall facilitate the operation, inspections, maintenance, repair and replacement of systems and components and shall limit the personnel exposure to ionizing radiation.
The buildings, compartments and components, which may be contaminated with radioactive substances, shall be designed in a way that allows easy decontamination by chemical or mechanical means.
The personnel access to compartments of high contamination level shall be controlled by means of locking devices with interlocks and indication for actuation and unavailability.
Biological protection shall be designed in a conservative way, taking into account the build-up of radio nuclides over the plant lifetime, the potential loss of shielding efficiency due to effects of interactions of neutron and gamma rays with the shielding, due to reactions with other materials, decontamination solution, and the expected temperature conditions in design basis accidents.
The choice of materials for the shield shall be made on the basis of the nature of the radiation, the shielding, mechanical and other properties of materials and space limitations.
Ventilation systems shall be installed to:

1. prevent spreading of gaseous radioactive substances in plant compartments;
2. reduce and maintain compartments' airborne concentrations below the established limits and as low as reasonably achievable in all operational states and design basis accidents;
3. cleanup the air in premises containing inert or harmful gases.

In designing a ventilation system, the following factors shall be taken into account:

1. mechanisms of thermal and mechanical mixing;
2. limited effectiveness of dilution in reducing airborne contamination;
3. exhausting of the air from areas of potential contamination at points near the source of contamination;
4. ensuring adequate distance between exhaust air discharge point and the intake point;
5. providing a higher pressure in the less contaminated zones in comparison with the zones of higher contamination level;
6. preventing the spread of fire-released smoke products to neighboring compartments.

Design shall provide for ventilation and air cleaning systems before discharge of gaseous radioactive substances to the environment.
Filters of air cleaning systems shall be sufficiently reliable to perform their function with the necessary decontamination factor in all operational modes. The design shall provide means to test their efficiency.
Provisions shall be made in the design for an automated system for radiation monitoring at the workplace and at the NPP site, and a system for radiation monitoring at the radiation protection and the monitored areas. These systems shall ensure the collection and processing of information on the radiation conditions, on the effectiveness of protective barriers, on the radionuclide activity, and information necessary to predict changes in the radiation conditions in all operational states and accident conditions.
The equipment of the automated system for radiation monitoring shall enable the implementation of:

1. radiation monitoring of technological environment
2. individual monitoring;
3. radiation monitoring at the workplace and at the NPP site;
4. area monitoring for limiting the spread of radioactive contamination.

The system of radiation monitoring shall be able to process and archive the data.
The laboratory methods and technical means of the system for radiation monitoring at the radiation protection and monitored areas shall ensure measurement of the content of human induced radio nuclides in soil, water, deposits, vegetation, water flora and fauna, and agricultural products.

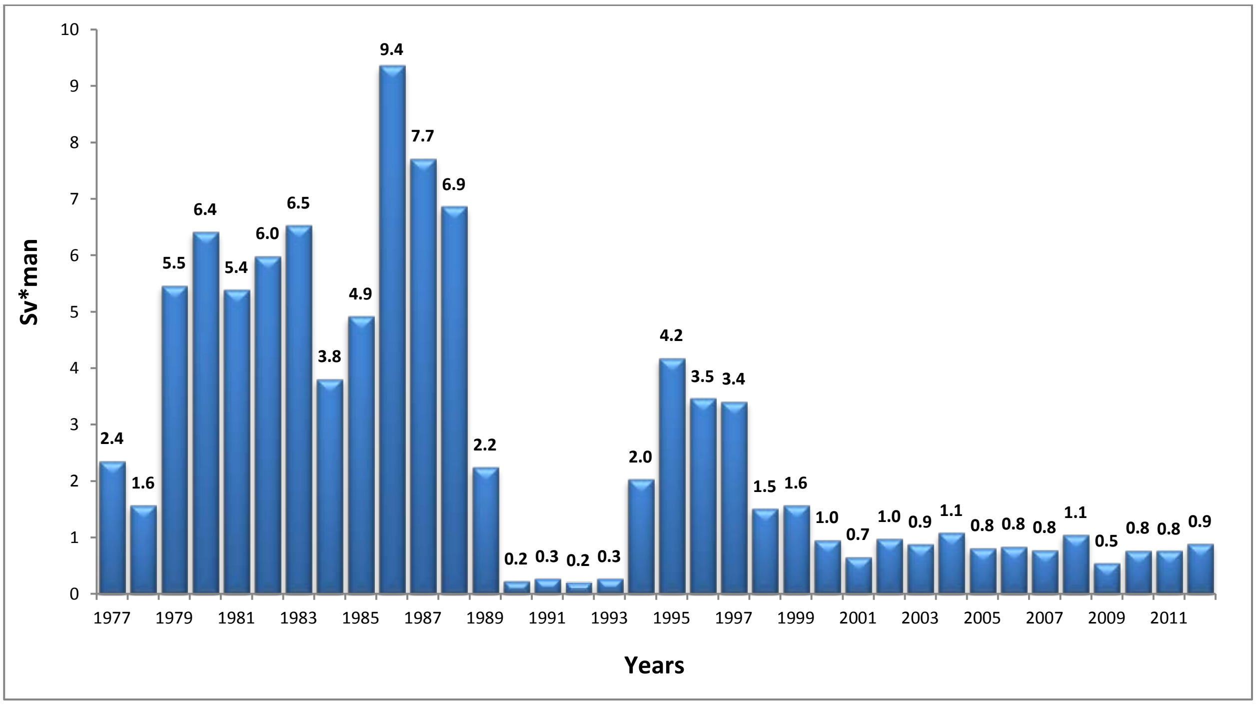
Radiation Protection at the Operation stage of Nuclear Power Plants

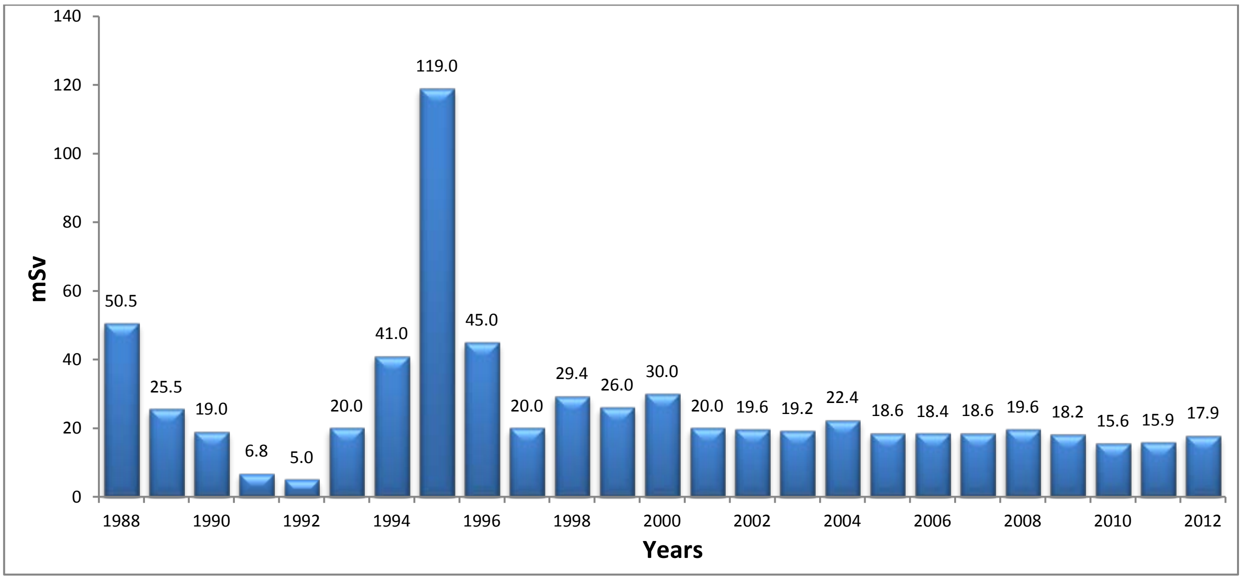
For the practical implementation of the radiation protection optimization at the Armenian NPP the ALARA Committee and ALARA Engineering group are formed. The ALARA Committee and the ALARA Engineering group work on continual basis in close contact with all the ANPP departments participating in activities with ionizing radiation sources, and implement activities on the Armenian NPP radiation protection optimization according to the requirements of the "Armenian NPP radiation protection management optimization according to ALARA principle" programme. Based on the results of the ALARA Committee activity report is prepared annually, which is the part of the Armenian NPP industrial activity annual report.

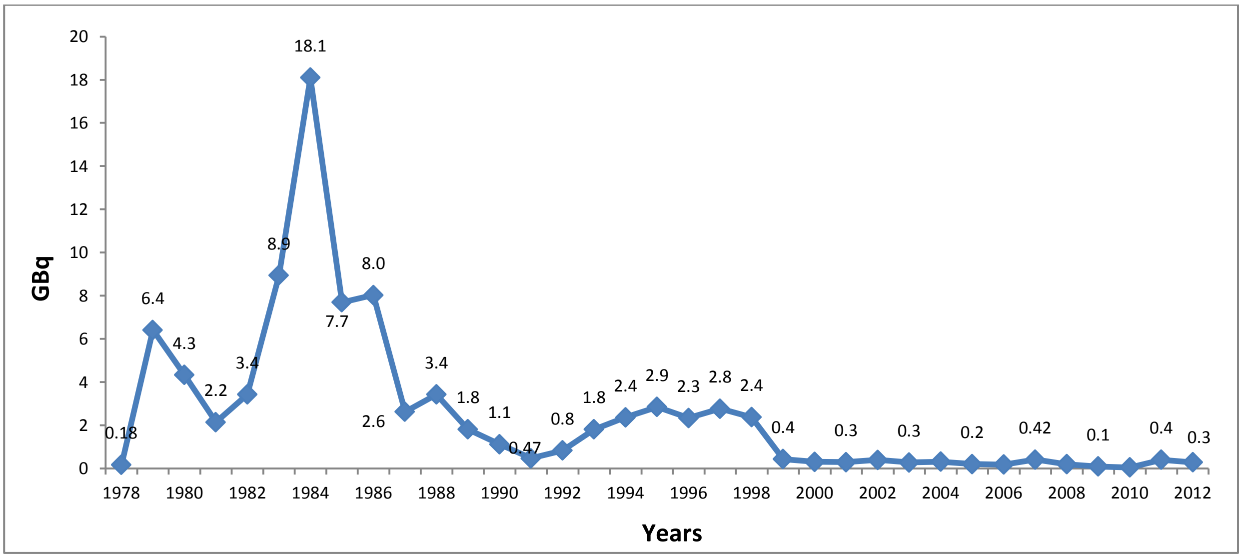
With the purpose of the ALARA principle further implementation at the Armenian NPP the "Program of the Armenian NPP Radiation protection" was developed which sets the objectives and tasks for minimization of the radiation impact and ensuring the effective radiation protection for the Armenian NPP personnel. It is aimed at maintaining the annual personnel collective dose rate as low as reasonably achievable.

The majority of the personnel annual effective dose consists of the doses received by the personnel implementing radiation hazard operations during the annual outage and refueling: non-destructive testing, decontamination works, repair works on systems and components. These activities are mainly performed by the Armenian NPP personnel and this explains the difference in doses received by the personnel and contractors.

The data on collective doses, individual maximum doses and the airborne releases are presented below.







The Regulatory Body review the submitted documents including this and other information.

Radiation Protection at the Decommissioning stage

The main approach for this stage is that all Radiation safety requirements for decommissioning of atomic energy utilization installations (including nuclear installations) should be provided in accordance with the requirements of the radiation safety regulations after complex radiation and technical studies of technological systems, equipment, building structures and adjacent territory of installation.

The main regulations for nuclear and radiation safety, the safety requirements, especially radiation protection requirements during preparation of Decommissioning plan of ANPP are the follows:

- The decommissioning plan shall include safety measures for different phases of decommissioning; shutdown, conservation, dismantling, re-use and disposal.
- Decommissioning program particularly shall include the time period and volume of works for the preliminary complex engineering – radiation investigations.
- Assessment of the total activity of spent fuel and the activity of separate radionuclide

- Assessment of the radioactivity of primary circuits and radioactivity of the main facilities, equipments, pipelines and engineering construction (both the activated and the surface contamination)
- Assessment of radiation contamination in premises of ANPP and the prognoses of changes in timeframe - up to design end time and final shutdown.
- The amount/volume activity of liquid and solid radioactive wastes (including radioactive bulk materials), the available treatment methodology and the characteristics of available storage and/ or disposal facilities.
- Assessment of the amount/volume of non-radioactive wastes (chemical, construction materials and others)
- Evaluation of the radiation situation at the beginning of decommissioning and the prognoses during each stages
- Assessment of projected /expected collective doses of personnel and public in each stages based on the radiation characteristics
- The real and expected liquid and airborne discharges
- Assessment of accrued unusual accidental situation during the commissioning of ANPP and their radiological consequences.

Based on these requirements the ANPP has prepared the first stage of radiological characterization of ANPP.

Radiological Characterization in Support of ANPP Decommissioning Scope and Objectives

The objective of radiological characterization is so provide a reliable database of information on quantity and type of radionuclides, their distribution and their physical and chemical states. Characterization involves a survey of existing data, calculations, in situ measurements and/or sampling and analyses. Using this database the decommissioning planner may assess various options and their consequences, considering:

- operating techniques: decontamination processes, dismantling procedures (hands on, semi-remote or fully remote working) and tools required;
- radiological protection of workers, general public and environment;
- waste classification;
- resulting costs.

Comparison and optimization of these factors will lead to the optimization of the process as a whole.

Data obtained during radiological characterization are extremely important for the work planning and corresponding decision making, taking into account that the decommissioning, as a rule, is a stage-by-stage process and each subsequent stage is planned and realized on the basis of the information received at earlier stages.