ISOE European Symposium
20-22 June 2012

Faculty of Nuclear Sciences and Physical Engineering

Prague, Czech Republic

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2012 ISOE EUROPEAN SYMPOSIUM

20-22 June 2012, Prague, Czech Republic

FINAL PROGRAMME
BOOK OF ABSTRACTS
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PROGRAMME OF THE SYMPOSIUM

WEDNESDAY 20 JUNE 2012

08:00 – 08:30	Registration
08:30 – 09:00	Opening Ceremony (ISOE, ČEZ, SÚJB)

Session 1. *Chairpersons*	**Regulatory Aspects** ***K. Petrova*** *(SÚJB, Czech Republic),* ***G. Abela*** *(EDF, France)*
09:00 – 09:15	Regulatory Approach to Radiation Protection in New NPPs *V. Riihiluoma (STUK, Finland)*
09:15 – 09:30	Radiation Safety as an Object of Licensing of New Nuclear Units in the Czech Republic *L. Urbančík (State Office for Nuclear Safety, Czech Republic)*
09:30 – 09:45	Dose Constraints and Other Policy and Practical issues in Occupational Radiation Protection *G. Frasch (BfS, Germany), M. Pinak (OECD/NEA, France), T. Lazo (OECD/NEA, France)*
09:45 – 10:00	Ongoing Efforts of HERCA on the Harmonisation of the Radiological Monitoring Systems for Outside Workers *A. Fremout (FANC, Belgium), I. Amor Calvo (CSN, Spain), B. Griciene (RSC, Lithuania), G. Frasch (BfS, Germany), R. Havukainen (STUK, Finland), M. Lehtinen (STUK, Finland), S. Léonard (FANC, Belgium), S. Mundigl (EC, Luxembourg), M. Nettleton (HSE, United Kingdom), M.-L. Perrin (ASN, France), I. Petkov (ABV, Bulgaria), M. Skarzewski (PAA, Poland), N. Svilicic (DZRNS, Croatia), C. Thijssen (SZW, Netherlands), S. Walker (HSE, United Kingdom)*
10:00 – 10:15	Definition and Access to Restricted Areas Regarding Radiation Protection Issues - The Innovative Expertise Process Launched by the French Safety Authority *N. Saâd, M.-L. Perrin, J.-L. Godet, O. Couasnon (ASN), C. Schieber (CEPN), France*
10:15 – 10:30	Full Session Discussion

10:30 – 11:15	*Coffee-break, Visit of Exhibition Booths, Posters*

WEDNESDAY 20 JUNE 2012

Session 2. *Chairpersons:*	**Accident Management** ***B. Le Guen*** *(EDF, France),* ***G. Renn*** *(Sizewell B NPP, United Kingdom)*
11:15 – 11:30	Status of the Fukushima NPP Accident *T. Linnemann (VGB PowerTech e.V., Germany)*
11:30 – 11:45	AREVA Experience in the Construction of the Water Treatment Facility *S. Bérard (AREVA NP, France)*
11:45 – 12:00	Reactions on Lessons Learned from Fukushima in Switzerland *S.G. Jahn (ENSI, Switzerland)*
12:00 – 12:15	Procedure for Dose Management at Forsmark NPP During Severe Conditions *S. Hennigor (Forsmark NPP, Sweden)*
12:15 – 12:30	Full Session Discussion

12:30 – 14:00	*Lunch*

Session 3. *Chairpersons:*	**RP Management** ***T. Zodiates*** *(EDF Energy, United Kingdom),* ***D. Fuchsová*** *(SÚJB, Czech Republic)*
14:00 – 14:15	Occupational Exposure in German Nuclear Power Plants *J. Kaulard, C. Schmidt, E. Strub (GRS), J. Junkersfeld (Federal Ministry for The Environment, Nature Conservation and Nuclear Safety), Germany*
14:15 – 14:30	Analysis of the Main Source-Term Reduction Efforts During the Planned Maintenance Works at Russian VVER (PWR)-1000 Reactors *V. Glasunov, Y. Yanchenko and others (VNIIAES, Russian Federation)*
14:30 – 14:45	Dose Reduction Actions for the Exposed Persons with Highest Individual Doses in EDF NPP *P. Le Genti, R. Masin, C. Marcillet (EDF UTO), M. Michelet (CEPN), A. Riedel (EDF UNIE/GPRE), France*
14:45 – 15:00	Inside and Outside - a Review of the RCA Boundary *I. Stalder, A. Ritter (Leibstadt NPP, Switzerland)*
15:00 – 15:15	Full Session Discussion

15:15 – 16:00	*Coffee-break, Visit of Exhibition Booths, Posters*

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Session 4.	**Source-Term Management**
Chairpersons:	***F. Kamping** (Borssele NPP, Netherlands), **B. Rosell Herrera** (Almaraz NPP, Spain)*
16:00 – 16:15	Radiation Protection Aspects of Water Chemistry and Source-Term Management with a View of an ISOE Expert Group *A. Rocher (EDF UNIE), L. Vaillant (CEPN), G. Ranchoux (EDF SEPTEN), H.B. Okyar (OECD/NEA), France*
16:15 – 16:30	Impact of Operational Events on Particulate Transport and Radiation Fields *D. Wells (EPRI, USA)*
16:30 – 16:45	EDF Source Term Reduction Project - Main Outcomes and Further Developments *G. Ranchoux, J. Bonnefon, M. Benfarah (EDF SEPTEN), M. Wintergerst, F. Gressier, P. Varry (EDF CEIDRE), S. Leclercq (EDF R&D), S. Blond (EDF DPN/EM), G. Cordier (EDF DPI), France*
16:45 – 17:00	BWR System Surface Contamination: Three Decades of Nuclide Specific Measurements *M. Olsson (Forsmark NPP, Sweden)*
17:00 – 17:15	CZT Technology Application at EDF for Better Radiation Protection and Nuclear Plant Surveillance *J. Bonnefon, G. Ranchoux, L. Guinard (EDF SEPTEN), A. Rocher (EDF UNIE), L. Piotrowski (EDF R&D), S. Blond (EDF DPN/EM), G. Cordier (EDF DPI), France*
17:15 – 17:30	EPR™ Reactor Activity Management: Design Performances and Chemistry Program *M.-H. Clinard, P. Jolivet, N. Engler, F. Chahma (AREVA NP, France)*
17:30 - 18:00	Full Session Discussion

18:30 – 19:30 **Prague City Walking Tour** along the Vltava River *(will bring you to the Conference Dinner)*

19:30 **Conference Dinner at Žofín Palace, sponsored by ČEZ**
Address: Slovanský ostrov 226, Praha 1 (metro B „Narodni trida“ + 10 min. by walk)

THURSDAY 21 JUNE 2012

08:00 – 08:30	Registration
Session 5. *Chairpersons:*	**Large Job Experiences** ***D. Steinel*** *(Philippsburg NPP, Germany),* ***O. Couasnon*** *(ASN, France)*
08:30 – 09:00	Ringhals 2 Fire in Containment & Lessons Learned *T. Svedberg (Ringhals NPP, Sweden)*
09:00 – 09:15	ALARA in Practice - Reactor Internal Parts Repair at the Temelín NPP *M. Hort (State Office for Nuclear Safety), J. Koc, O. Kvasnička (Temelin NPP), P. Zapletal (Škoda JS), Czech Republic*
09:15 – 09:30	Operational Experience of the Replacement of Pressuriser Heaters during a Forced and a Planned Refuelling Outage *M. Lunn, G. Renn (Sizewell B NPP, UK)*
09:30 – 09:45	Primary System Decontamination of the German PWR Grafenrheinfeld - Process Application and Recontamination Experience *B. Stellwag (AREVA NP GmbH), S. Schütz, A. Jacob (Grafenrheinfeld NPP), C. Stiepani, C. Topf (AREVA NP GmbH), Germany*
09:45 – 10:00	System Decontamination of Two BWRs *J. Brunk (Forsmark NPP, Sweden)*
10:00 – 10:15	Full Session Discussion

10:15 – 11:00	*Coffee-break, Visit of Exhibition Booths, Posters*

Session 6. *Chairpersons:*	**Instrumentation and Measurements** ***S. Hennigor*** *(Forsmark NPP, Sweden),* ***L. Jentjens*** *(VGB PowerTech e.V., Germany)*
11:00 – 11:15	An Overview of Electricité de France Research and Development Projects in the Field of Occupational Radiological Protection *E. Gaillard-Lecanu, S. Jahan, R. Kutschera, I. Fucks, E. Courageot (EDF R&D, France), H. Mohand-Kaci (CS), G. Cordier (EDF DPN), France*
11:15 – 11:30	Underwater Diving Remote Monitoring Implementation *M. Leasure (Braidwood NPP, USA) - 2012 Ft. Lauderdale distinguished paper*
11:30 – 11:45	Lowering the Radioactive Aerosol Emission from Forsmark 1, 2 and 3 *R. Schmocker, T. Björkman (Forsmark NPP, Sweden)*
11:45 – 12:00	Research and Development of Gas and Aerosols Dispersion in the RB at EDF Nuclear Power Plants *M. Lestang (EDF UNIE), H. Mohand-Kaci (CS), J. Fazileabasse (EDF UNIE), L. Ricciardi, C. Prevost, L. Bouilloux (IRSN), S. Jahan (EDF R&D), G. Cordier (EDF DPN), France*

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12:00 – 12:15	Supervision of Portable Ventilation Unit (PVU): a Way of Defense to Control the Habitability of the Reactor Building during Outage in EDF Nuclear Power Plants *J.E. Maurer (Tricastin NPP), P. Bignon, G. Foy (Delta Neu), M. Lestang (EDF UNIE), H. Salmon, A. Sevenier (Tricastin NPP), P. Oleksy (EDF UTO), S. Marques (EDF UNIE), M. Boissout (Tricastin NPP), France*
12:15 – 12:30	Full Session Discussion

12:30 – 14:00	*Lunch*

Session 7. *Chairpersons:*	**Decommissioning** ***T. Svedberg** (Ringhals NPP, Sweden), **T. Taylor** (Mühleberg NPP, Switzerland)*
14:00 – 14:15	Radiation Protection during Decommissioning of Nuclear Facilities - Experiences and Challenges *J. Kaulard, C. Schmidt, E. Strub (GRS), Germany*
14:15 – 14:30	Radiation Protection at Decommissioning Stage *R. Buckermann (AREVA NP GmbH, Germany)*
14:30 – 14:45	Shovel Monitor for Sorting of Loose Decommissioning Materials from NPP *A. Slaninka, O. Slávik, M. Lištjak (VUJE, Slovakia)*
14:45 – 15:00	Radiological Characterization of V1 NPP Technological Systems and Buildings - Activation *J. Svitek (JAVYS NPP), K. Krištofová, T. Rapant (AMEC Nuclear Slovakia s.r.o.), Slovakia*
15:00 – 15:15	Radiation Area Classification and Sizing of the Storage buildings of used-up steam generators - Practical application *T. Canal, J. Routtier, X. Michoux (EDF CIPN, France)*
15:15 – 15:30	Solidification of Spent Ion Exchange Resins into the SIAL® Matrix at the Dukovany NPP *M. Pražská, P. Tatransky (Amec Nuclear Slovakia s.r.o.), P. Kopecký (CES a.s.) , Czech Republic*
15:30 – 15:45	Full Session Discussion

15:45 – 16:30	*Coffee-break, Visit of Exhibition Booths, Posters*

16:30 – 16:45	Conclusions from Radiation Protection Managers and Regulatory Body Meetings

THURSDAY 21 JUNE 2012

Session 8. *Chairpersons:*	**Internal Contamination Management** ***A. Ritter*** *(Leibstadt NPP, Switzerland),* ***M. Fárníková*** *(Temelin NPP, Czech Republic)*
16:45 – 17:00	EPRI Alpha Guideline Update *P. Tran (EPRI, USA)*
17:00 – 17:15	Internal Dosimetry Monitoring: A Comparative Study of the Effectiveness of Bioassay, Nose-Blow and Personal Air Sampling Measurements *R. K. Bull, G.A. Roberts, R. Talbot (Nuvia Ltd, UK)*
17:15 – 17:30	Occupational Exposure in a CANDU NPP: Management of the Risk for Internal Alpha Contamination *C. Chitu, I. Popescu, M. Baraitaru, A. Nedelcu, V. Simionov (Cernavoda NPP, Romania)*
17:30 – 17:45	Tritium in German LWRs - Relevance on Radiation Protection *W. Schwarz (formely Isar NPP, Germany)*
17:45 – 18:00	Full Session Discussion

18:00 – 18:15	Distinguished Papers and Closure of the Symposium

20:00 – 23:00 **Celebration of the 20th Anniversary of ISOE System:**
Boat Tour with Dinner on the Classic River boat

FRIDAY 22 JUNE 2012

Visit of Nuclear Power Plants	
08:00	Bus Departure to Temelin NPP Bus Departure to Dukovany NPP
	Visit of Temelin Nuclear Power Plant Visit of Dukovany Nuclear Power Plant
17:00	Return to Prague (arrival approx. at 20:00)

Note: The participants to the NPP visit are invited to schedule their return flight on Saturday.

SOCIAL EVENTS

Costs of Social Events are included in Registration Fees.

Walking Tour along the Vltava River – Wednesday 20 June – 18:30-19:30

This short walking tour will start at the Faculty of Nuclear Engineering and will lead you along the embankment of the river Vltava with panoramic view of the Prague Castle and Lesser Town. You will pass along the picturesque Gothic Charles Bridge with a mighty Old Town tower, the monumental building of the Klementinum – University Library, St. Salvator church and Knights of the Cross church. Shortly you will get to the historical building of the National Theatre which was erected in the 19th century and finally to the Zofin Island with Žofín Palace which has been a leading centre of cultural and social life in Prague since 1837. Includes: English-speaking guide

Conference Dinner at Žofín Palace – Wednesday 20 June – 19:30

In 1835, when Václav Novotný decided to build the first brick building on Slavonic Island (or Dyers' Island as it was then called), he had no idea that the one-storey neoclassical building named after the mother of Emperor Franz Joseph I, Princess Sophie (or Žofie in Czech), would become one of the most important cultural and social centres in Prague and the Czech Lands as a whole. In 1884 the municipality bought the whole island, including Žofín, and over the next two years it converted the building into the present neo-renaissance palace with richly decorated interiors and fine halls.

Celebration of the 20th Anniversary of ISOE System: Boat Tour with Dinner on the Classic River Boat – Thursday 21 June – 20:00-23:00

An excellent opportunity to relax on fresh air, listen to nice music and enjoy delicious meals while admiring the beauties of the Prague city centre. Embark one of the boats cruising on the Vltava river just round the city centre, passing by the main sights. You will sail under the Charles Bridge with enormous open air gallery of Baroque statues and along the amazing Panorama of Prague Castle, romantic Kampa Island as well as other remarkable sights such as National Theatre, Vysehrad legendary hill etc. During the cruise you will listen to the pleasant music playing popular melodies and the guide´s description on the main sights passing by and the Prague history in general.

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REGULATORY APPROACH TO RADIATION PROTECTION IN NEW NPPs

Veli RIIHILUOMA

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Radiation and Nuclear Safety Authority (STUK) reviewed the utility Teollisuuden Voima Oyj's (TVO) application for the Construction Licence of the Olkiluoto 3 nuclear power plant unit in 2004 – 2005. Based on this review STUK prepared its statement on safety together with a safety assessment report of the new plant to the Government.

STUK has continued reviewing the detailed design documents of different systems of the new plant unit. STUK has also carried out periodic inspections during construction phase. The paper will discuss some radiation safety related requirements in the design of a new Finnish NPP and their implementation in the licensing documentation. Also some practical issues in review process are dealt with. There are a lot of challenges but also opportunities in radiation protection. The operation license process of Olkiluoto 3 will bring new aspects for reviewing process.

For the moment Olkiluoto 4 and Fennovoima 1 reactor projects are preparing readiness for construction licence applications.

By virtue of the Nuclear Energy Act (990/87) and the Government Decree on the Safety of Nuclear Power Plants (733/2008), STUK issues detailed regulations, YVL Guides, concerning the safety of nuclear power plants. Several YVL Guides deal with radiation safety essential in design and in construction (site, abatement of releases, worker radiation protection, water chemistry, emergency arrangements, etc.). At the moment a large project is going on to up-date all YVL-Guides taking in account the experiences in Olkiluoto 3 licensing process.

RADIATION SAFETY AS AN OBJECT OF LICENSING OF NEW NUCLEAR UNITS IN THE CZECH REPUBLIC

Libor URBANČÍK

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The Czech Republic plans to build new nuclear units in localities of the existing nuclear power stations at the Temelin Nuclear Power Plant and at the Dukovany Nuclear Power Plant (hereinafter referred to as the “NPP”). These new units should be built up to the end of the twenties. That planning is linked with an invitation to a tender and licensing. This paper describes the Czech way in the licensing process of the new nuclear units.

The State Office for Nuclear Safety (hereinafter referred to as the “SUJB”) is responsible for licensing and regulating the operation of both commercial nuclear power plants and research reactors in the Czech Republic. The Czech Republic, namely SUJB, has vast experience with licensing, having undertaken that process for Temelin NPP 1, 2 in the late nineties. Now, it is planning to build Temelin NPP 3, 4 and Dukovany NPP 5.

The Czech licensing process should be done according to the Czech legislation framework. There is sophisticated legislation in the field of radiation protection at nuclear stations in the Czech Republic. However, this legislation is acceptable for licensing process use, and that has been examined in practice, the Czech approach does not exclude the application of a broader and better worked-out licensing procedure, using foreign experience, for instance.

Usually, the licensing process has three levels: permit for placement, the preliminary safety analysis report (hereinafter referred to as the “PSAR”) and the final safety analysis report (hereinafter referred to as the “FSAR”). Czech legislation requires five levels, as follows:

- Environmental Impact Assessment documentation,
- Initiative safety analysis report,
- Preliminary safety analysis report,
- Pre-operational safety analysis report
- Final safety analysis report (i.e. operational safety analysis report)

This paper deals with particular items of the licensing process, as mentioned above, and analyzes issues connected with it. Requirements for safety analysis reports (hereinafter referred to as the “SAR”) from the SUJB to a licensee are processed to a database, which is strongly hierarchically organized. Requirements from the Czech Atomic Act take place on the top level of the pyramidal structure. Below this layer is level occupied by applications of legal regulations (decrees) specifying legal requirements more in details. A further level is fixed for requirements from IAEA documentation, WENRA recommendations and EUR documentation. The bottom level of that pyramidal database is determined for very detailed requirements carried out in accordance with both Czech and international standards.

DOSE CONSTRAINTS AND OTHER POLICY AND PRACTICAL ISSUES IN OCCUPATIONAL RADIATION PROTECTION

G. FRASCH (BfS, Germany), M. PINAK (OECD/NEA, France), T. LAZO (OECD/NEA, France)

The Expert Group on Occupational Exposure (EGOE) acts on behalf of the Committee on Radiation Protection and Public Health (CRPPH) of the OECD Nuclear Energy Agency. One of its mandates is to analyse the implementing of radiation protection principles into occupational radiation protection frameworks of workers. In particular, it focuses on the concept of dose constraints and some other policy and practical issues in occupational radiation protection in the nuclear sector.

ICRP Publication 103 regards the concept of dose constraints as a key factor in optimisation. In occupational exposure, dose constraints are demanded as individual, source related upper bounds below which exposures should be kept as low as reasonably achievable. This concept has also been adopted in the revised Basic Safety Standards of the IAEA and the European BSS.

EGOE analysed the perception and practical implementation of the dose constraint concept in several member states of the OECD. It also performed a survey on the use of dose constraints as a tool for optimisation of occupational radiation protection in the nuclear and parts of the non-nuclear sector. The report which was approved for publication by the CRPPH revealed that dose constraints are widely used, albeit often in a far more complex way then proposed by ICRP. Many, mainly European countries, follow the concept very closely. In the USA, constraints are used if appropriate and embedded in or subordinated to a total risk management. Japan, however, prefer to seek target doses below their national dose limits.
The dose constraint concept raises many open questions during the practical implementation process and requires guidance and close communication between regulators, licensees, and radiation protection managers.

Under the aspect of globalisation and worldwide workforce transfer, EGOE identified several policy and practical issues that pose a challenge to future occupational radiation protection:

- policy making: same international requirements lead often to different national implementations and practices,
- knowledge transfer and education between generations and nations,
- itinerant workers: implications of different national dose limits, transnational dose tracking and balancing, international radiation passbook,
- Management of high radiation risk jobs and jobs with combined workplace risks (radiation, toxic, chemical, ...),
- International information exchange on good practices and lessons learnt.

Based on the mandate and approval of the CRPPH, the expert group will work on these topics during the next two years to prepare another report.

ONGOING EFFORTS OF HERCA ON THE HARMONISATION OF THE RADIOLOGICAL MONITORING SYSTEMS FOR OUTSIDE WORKERS

Fremout, An, FANC, Belgium, an.fremout@fanc.fgov.be (Presenting); Amor Calvo, Ignacio, CSN, Spain; Griciene, Birute, RSC, Lithuania; Frasch, Gerhard, BfS, Germany; Havukainen, Ritva, STUK, Finland; Lehtinen, Maaret, STUK, Finland; Léonard, Sophie, FANC, Belgium; Mundigl, Stefan, EC, Luxembourg; Nettleton, Michael, HSE, United Kingdom; Perrin, Marie-Line, ASN, France; Petkov, Ivaylo, ABV, Bulgaria; Skarzewski, Maciej, PAA, Poland; Svilicic, Niksa, DZRNS, Croatia; Thijssen, Carel, SZW, Netherlands; Walker, Steve TD, HSE, United Kingdom

The association HERCA brings together the Heads of European Radiological protection Competent Authorities. It consists of a Board of Heads and topical working groups composed of experts from the different Radiation Protection Authorities.

Its objectives are to build and maintain a network of chief radiation safety regulators in Europe, which promotes exchange of experience, where appropriate, expresses its consensus opinion on significant regulatory issues, and develops, by consensus whenever possible, a common approach to radiological protection issues within the States of HERCA members. The association involves, as appropriate, the European Commission and other relevant stakeholders in its activities.

HERCA Working Group 1 (WG1) was created in 2007 to investigate on the practical implementation of the Directive 90/641/Euratom and on how a better harmonisation of the radioprotection systems for outside workers could be achieved. In 2008, a survey was lead about the practical transposition of the Directive within the Member States. It allowed to derive the commonalities and variations of the radiation monitoring systems for outside workers and to compare the content registered in the radiation passbooks to the required information in the Directive. A model of radiological passbook was proposed by WG1, including the harmonisation of terminology and of the requirements on data content of a Radiation Passbook, with a distinction between mandatory fields and optional fields. The Radiation Passbook can be a paper based system but countries could also opt to use an electronic (possibly web-based) system instead of parts of the paper based system.

After approval of the proposal by the Board of Directors, it was sent to the European Commission for its inclusion in the BSS recast. A proposal of inclusion in a generic way has been formulated. Additionally, HERCA invited all European national competent authorities and stakeholders to express their comments.
In 2010, WG1 has been given the new mandate to carry out a feasibility study for the transition to electronic information exchange between countries to guarantee a protection of cross-boarder outside workers equivalent to the protection of other workers. This work should be carried out in close collaboration with the EC and with ESOREX. Starting from 2012, the EC can take this feasibility study as a starting point to look more closely to the technical solutions for such an electronic information exchange system.

**DEFINITION AND ACCESS TO RESTRICTED AREAS REGARDING RADIATION PROTECTION ISSUES
THE INNOVATIVE EXPERTISE PROCESS LAUNCHED BY THE FRENCH SAFETY AUTHORITY**

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As European requirements for radiation protection are about to get updated (future Council directive laying down basic safety standards for protection against the dangers arising from the exposure to ionising radiation), the French Safety Authority (ASN) and the General Directorate of Labour (DGT) consulted, in February 2011, the ASN Advisory Committees[1] (GPRAD and GPMED) in order to have their opinions and recommendations concerning the definition and access to restricted areas regarding radiation protection issues.

This consulting is part of a prospective reflection process of the ASN Advisory Committees. Beyond the critical analysis of the existing regulatory framework, GPRAD and GPMED experts will have to take into account future national and European requirements. In this context, they will have to propose one or several regulatory systems in order to define and manage the access to restricted areas.

A task group, « GT-Zonage », has been created among the ASN Advisory Committees, in particular to elaborate a survey of national and international operational practices. The actions carried out by the « GT-Zonage » consisted, for instance, in spreading a feedback form designed for operational actors in every activity sectors (medical, industrial, nuclear, transport…) or in organizing a national seminary that brought together about 200 persons and focused on the specific matter of the definition and the access to restricted areas.

The « GT-Zonage » also required the support of an external contractor (CEPN) in order to use the conclusions of a wide study of international practices regarding restricted areas. This specific study will be presented in this symposium.

In 2012 (first semester), GPRAD and GPMED will express their opinions based on the analysis carried out by the « GT-Zonage ». Then, ASN and DGT will take position on these recommendations.

[1] Seven Advisory Committees have been created and report to the ASN General Director.

PROCEDURE FOR DOSE MANAGEMENT AT FORSMARK NPP DURING SEVERE CONDITIONS

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This presentation will explain how dose management is applied during severe conditions/accidents at Forsmark NPP. As an example from these procedures the following dose constraints will be used:

Category	Aim of the operation	Dose constraint
1	- Prevention of loss of human life	> 100 mSv All reasonable protective measures shall be undertaken in order to keep the dose below 250 mSv
2	- Prevention of serious core damage - Prevention of major releases of radioactive specimens	< 100 mSv
3	- Prevention of serious human injuries - Prevention of large collective doses - Prevention of serious expansion of the accident progression (for example the operation of mitigating systems) - Recovery of reactor safety systems	< 50 mSv
4	- Short term recovery measures - Urgent protective measures - Sampling and monitoring in the environment	< 20 mSv
5	- Long term recovery measures - Measures not directly connected to the emergency situation	Dose limits for normal operation shall be followed, taken into account the ALARA principle

In the light of the Fukushima accident also some questions and problems regarding operative dose management will be raised and discussed.

OCCUPATIONAL EXPOSURE IN GERMAN NUCLEAR POWER PLANTS

Jörg KAULARD, Claudia SCHMIDT, Erik STRUB[1)]
Jörg JUNKERSFELD[2)]

[1)] Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS) mbH, Germany
[2)] Federal Ministry for The Environment, Nature Conservation and Nuclear Safety, Germany

As of January 2012, 9 nuclear power plants are in operation and will be shut down stepwise until 2022. Further 8 nuclear power plants are finally shut down and 16 nuclear power plants are under decommissioning.

Within this contribution an overview will be provided on the occupational exposure in the German nuclear power plants in operation and under decommissioning. The contribution will explain the current trends in occupational exposure and will highlight specific aspects contributing to these trends. Among others the occupational radiation protection in German nuclear power plants is based on the European Radiation Protection directive 96/29/EURATOM. This directive will be replaced by the new European Basic Safety Standards, which have been proposed by the European Commission and are discussed in the Council Working Party. This contribution to the ISOE Symposium will provide a first, very preliminary outlook on some potential consequences to the management of occupational exposure in German nuclear power plants. E.g. the changes expected due to regulations regarding the status of outside workers, the exchange of radiological data of workers and the dose limit of the eyes will be discussed.

ANALYSIS OF THE MAIN SOURCE-TERM REDUCTION EFFORTS DURING THE PLANNED MAINTENANCE WORKS AT RUSSIAN VVER (PWR)-1000 REACTORS

Vadim Glasunov [1], Yury Yanchenko [1] and others

[1] All-Russian Research Institute for NPP Operation (VNIIAES)
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At Russian VVER (PWR)-1000 MWe reactors, from 80 to 90% of the annual collective doses resulted from the outages. According to this reason, special attention was paid on source-term reduction management during implementation of planned maintenance works to meet the common aim of utilities employees and contractors dose optimization.

The paper presents:

- evolution of annual and outage collective doses for VVER (PWR)-1000 MWe units;
- evolution of three-year rolling average annual and outage collective doses;
- main factors influenced on decrease of occupational exposure indicators;
- three-year rolling average collective dose per the most important systems and jobs;
- methods used to decrease occupational exposure indicators for different systems and jobs.

DOSE REDUCTION ACTIONS FOR THE EXPOSED PERSONS WITH HIGHEST INDIVIDUAL DOSES IN EDF NPP

Pascal LE GENTI[1], Roland MASIN[1], Clément MARCILLET[1], Marie MICHELET[2], Alexandre RIEDEL[3]

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[2] CEPN, 28 rue de la Redoute, 92260 Fontenay-aux-Roses
[3] EDF/DPN/UNIE/GPRE, Cap Ampère, 1 Place Pleyel, 93282 Saint Denis Cedex

Proposal for oral presentation in 2 parts

1. Analysis of the individual doses taken at EDF NPPs.

The study consists in analyzing the database of every dose movements recorded in the French NPPs between January 2006 and December 2009. A dose movement corresponds to the dose taken by one person between access to and exit from a radiation controlled area. Apart from the dose, the database indicates the worker's trade, his company, the date, the duration of the stay in the radiation controlled area, the task code, etc. This database allows us to focus on the individuals whose 12-rolling-month dose was greater than 10 mSv, their trade, company and the tasks leading to high dose.

The main jobs leading to high doses are steam generators and reactor vessel maintenance. In particular, the tasks leading to high dose around the reactor vessel are the opening, the closure of the reactor vessel and all the maintenance between.

The main tasks around the reactor vessel are the following:
- Decontamination of the reactor cavity, the spent fuel pit, etc.
- Opening and closure of the reactor vessel,
- Reactor vessel and fuel closed circuit TV inspection,
- Maintenance of the tapping of the reactor vessel,
- Repairing of the tapping of the reactor vessel,
- Fuel handling system,
- Reactor building handling equipments.

2. Analysis of the activities and dose reduction actions

After this analysis, EDF and CEPN have met the main contractors in charge of these operations to identify the tasks leading to high doses and good practices and improvements to reduce the dose in terms of techniques, organization, process, etc. that are not yet implemented in all French NPPs.

For example, one specific task on the reactor vessel, which leads to high dose, is the mounting/demounting of thermocouples. For this task, the contractors intervening on the NPP, have been working on a new system of thermocouple waterproofness faster to install than the previous one, which reduce the individual dose associated with this task.

After these meetings, a guide was written with some questions for NPPs about RP organization, assessment of 'identified good practices'. This guide was tested in 2010 on 4 NPPs and was sent to the others in 2011.

The main objective was to identify potential actions, which could be taken to reduce doses received on these jobs, which lead to high individual doses. The main actions will be presented.

INSIDE AND OUTSIDE – A REVIEW OF THE RCA BOUNDARY

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After some KKL-events with releases of radioactivity to the environment through unlicensed pathways, the regulator required KKL to review the entire boundary of the Radiation Controlled Area. Most of the findings were unrecognised and often undocumented "holes" originating from the building phase of the plant, such as abandoned tubes for electric cables. Some were simply due to bad design back in the 1970s. A special problem encountered was pipes carrying non-radioactive materials like rainwater drains crossing the RCA. Numerous findings resulted in many plant modifications, some of them major including a complete new building for the cold condensate storage tanks. Based on the experience in KKL, the regulator requires similar examination and renewal of the boundary of the RCA in the other plants in Switzerland.

RADIATION PROTECTION ASPECTS OF WATER CHEMISTRY AND SOURCE-TERM MANAGEMENT WITH A VIEW OF AN ISOE EXPERT GROUP

A. Rocher (EDF, France), L.Vaillant (CEPN, France), G. Ranchoux (EDF, France),
H.B. Okyar (OECD/NEA, France)

Information System on Occupational Exposure (ISOE) provides a forum for radiation protection professionals from nuclear electricity utilities and national regulatory authorities worldwide to share dose management information and operational experience to improve the optimisation of worker radiological protection at nuclear power plants. The ISOE programme, with its participating utilities and regulatory authorities, is a key organisation in developing safe, sustainable and socially acceptable strategies for emerging issues in the field of occupational radiation protection.

Water chemistry approaches in different design of NPPs vary in results and consequences in terms of radiation protection performance. It was suggested that radiation protection aspects of primary system water chemistry and source-term management could be discussed in detail by the participation of ISOE utilities with an establishment of an expert group. The Expert Group on Water Chemistry and Source-Term Management (EGWC) has been mandated to address the experience of various ISOE utilities with various water chemistry regimes to explore if experience exchange could help to improve radiation protection performances. It is also necessary to note that water chemistry should not be viewed only from the context of radiation protection issues, but also from the context of operational and safety issues, and it was proposed to be grouped into a few of the most commonly used water chemistry approaches (e.g. zinc injection, pH control, iron injection, hydrogen water chemistry, etc.) to focus the exchange of experience discussions.

The expert group is planning to focus on:

- Description of strategies and techniques aiming to limit the level of activity in the primary coolant (prevention of contamination);
- Description of strategies and techniques for the decrease of activity in the primary coolant or circuit decontamination (remediation of contamination);
- Performance indicators to assess results from the above strategies and techniques; measurement techniques and performance assessment (monitoring), and
- Management of iodine, xenon and alpha risks.

The outcome of the work will be a new ISOE publication which is planned to include information and practical experience available in the nuclear industry on addressing operational aspects of primary water chemistry and source-term management of nuclear reactors with special emphasis on effects on the management of occupational exposures, identify factors and aspects which play key roles in achieving good practices in water chemistry management and analysis on impact on worker doses and operational costs.

IMPACT OF POWER UPRATES AND OPERATIONAL EVENTS ON PARTICULATE TRANSPORT AND RADIATION FIELDS

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Recent changes in core design and plant operation have impacted radiation fields in various systems throughout BWR and PWR plants. It has been proposed that unexpected variations in radiation fields have been related to particulate activity deposition, particularly late in the cycle and during shutdowns. These unexpected variations are likely related to the observed disconnect between standard piping dose rates and cumulative exposure variations and trends. This is particularly the case in auxiliary systems where the collection of dose rate data is not standardized. These auxiliary radiation field measurements must be evaluated in order to provide sufficient information about shutdown dose rate, particularly during transients, to validate this hypothesis on a generic basis.

This presentation will include analysis of recent trends in radiation field variations at selected BWRs and PWRs. The impacts of core design, operations and chemistry, i.e., especially particulate transport, on surface activity concentrations on reactor coolant and auxiliary system piping and components will be discussed.

EDF SOURCE TERM REDUCTION PROJET - MAIN OUTCOMES AND FURTHER DEVELOPMENTS

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The dose reduction is a strategic purpose for EDF in link with the stakes of, nuclear acceptability, respect of regulation and productivity gains. This consists not only in improving the reactor shutdown organization (time spent in control area, biological shielding,…) but also in improving the radiological state of the unit and the efficiency of the source term reduction operations.

Since 2003, EDF has been running an innovative project called “Source Term Reduction” federating the different EDF research and engineering centers in order to:

- participate to the long term view about Radiological Protection issues (international feedback analyses),
- develop contamination prediction tools (OSCAR software) suitable for the industrial needs (operating units and EPR design),
- develop scientific models useful for the understanding of contamination mechanisms to support the strategic decision processes,
- carry on with updating and analyzing of contamination measurements feedback in corrosion products (EMECC and CZT campaigns),
- carry on with the operational support at short or middle term by optimizing startup and shutdown processes, pre-oxydation or and by improving purification efficiency or material characteristics.

This paper will show in a first part the main 2011 results in occupational exposure (collective and individual dose, RCS index …). In a second part, an overview of the main EDF outcomes of the last 3 years in the field of source term reduction will be presented. Future developments extended to contamination issues in EDF NPPs will be also pointed out in this paper.

BWR SYSTEM SURFACE CONTAMINATION: THREE DECADES OF NUCLIDE SPECIFIC MEASUREMENTS

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The Swedish BWR fleet was built and taken into operation during the period 1972–1985. All of the nine units that were built were delivered by ASEA Atom who, as a part of the suggested operational follow-up, deviced a methodology to measure the nuclide specific surface activity inside pipes and heat exchangers. A low efficiency HPGe detector is shielded, collimated and used for such measurements. The efficiency of each measurement is then calculated, and results are evaluated (Bq/m2).

At the Forsmark NPP the surface contamination measurements have been done, during the annual outages of the three units of the site, for the last 30 years. The long term trends can be used to evaluate the sources of system contamination, which may also give hints on the causes of contamination. As the cause of a contamination is known, it may be easier to take the proper steps to control the source term and to reduce contamination and dose rates. The surface contamination measurements thus become a tool for ALARA work, as well as a part of the long-term follow-up of the plant chemistry status.

About a dozen measurement points are included in a typical measurement campaign during an outage:

- On each of the two pipes that lead water from the reactor tank to the shutdown cooling system. This system also feeds water to the reactor water cleanup system (RWCU).
- On the pipes and selected heat exchangers along the the RWCU system: before heat removal, after heat removal but before filtration, after filtration and after final regenerative heat exchange. The temperature dependence of the surface contamination can be shown.
- On a pipe for the system that supplies water to the hydrualic scram function and to the crud removal flow through the control rod guide tubes. This water is a partial flow of the filtrated water from the RWCU system.
- On a pipe of the cooling and clean-up system for the fuel pool water.
- On two of the steam lines close to the high pressure turbine.

A number of examples have been compiled to show long term trends and some of the cause and effect relationships that have been seen through the years. The examples include the strong connection between moisture in fresh steam (carry-over) and contamination of steam pipes, and also the possible effect of fuel spacer corrosion.

CZT TECHNOLOGY APPLICATION AT EDF FOR BETTER RADIOPROTECTION AND NUCLEAR PLANT SURVEILLANCE

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Dose reduction during nuclear power plant (NPP) outages is a strategic objective for EDF (working conditions improvement, nuclear acceptability, respect of regulation and productivity gains). Obtaining this aim is correlated to source term reduction and, thus, contamination characterization and dose rate measurements.

In order to study the impact of chemistry, operation and plant design parameters on radiation fields, EDF has put in place a global monitoring program including in particular volume activity monitoring, dose rate measurements and gamma spectrometry acquisitions. Consequently, for over 40 years, EDF has performed gamma spectrometry measurements with CEA (EMECC system using a Ge detector), with more than 300 campaigns in EDF plants and also about 70 campaigns in collaboration with other nuclear operators in Europe.

In addition to EMECC campaigns and dose rate measurements (RCS and reactor Building Indexes), EDF has been using a portable gamma spectrometer with interchangeable CZT (Cd-Zn-Te) probes to implement a new spectrometric measurement program since 2006. These CZT probes function at ambient temperatures (semi-conductor technology) and allow one to acquire gamma spectra over the energy range 300 – 1800 keV for equivalent dose rates varying from 0.1 to 500 mSv/h. EDF usually uses an uncollimated probe that is placed directly on contact with the circuit component.

All EDF NPP plants (radioprotection services) are equipped with this portable CZT gamma spectrometer and special analysis software.

Comparison with EMECC results, which are more accurate but also more difficult to handle, shows that the EDF CZT spectrometer is able to satisfactorily (i) identify the dominant corrosion and fission products present within radioactive contaminations, and (ii) quantify their relative contributions to the total equivalent dose rate.

This communication will first summarize the main characteristics of the EDF CZT spectrometer (acquisitions and analysis). In the second part, the recent feedback between 2006 and 2011, taking into account the new updated measurement program, will be presented. A statistical analysis including the general trends for each series, and the observed impact on radioactive contaminations when replacing steam generators will also be described.

Furthermore, the CZT positioning in the EDF measurement strategy will be discussed and some practical applications rolled out on EDF fleet will be described. CZT gamma spectrometry is a very useful tool for the evaluation of radiological situations present within an NPP, for following NPP evolution (new pollutions) and for comparing a given NPP with other plants.

Finally, new developments in progress or planned will be presented (feasibility study for surface activity measurements, characterization of filter retention performances).

EPR™ REACTOR ACTIVITY MANAGEMENT: DESIGN PERFORMANCES AND CHEMISTRY PROGRAM

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The ALARP approach has been applied for determining the EPR™ Reactor Activity Management through the different phases of design and operation.

The process developed by AREVA is based on the balance between the theoretical developments, the laboratory tests and the NPP experience. The identification and the characterization of the source term constitute the basis for making decision process concerning the material choice, the system design provisions, the primary coolant chemistry conditioning and the chemistry/radiochemistry monitoring.

The application of this methodology provides qualitative and quantitative assessments explaining that the design, sizing and chemistry conditioning of EPR™ primary circuit are adapted to supply the correct activity management.

The methodology developed by AREVA reveals the EPR™ reliability and availability for preventing and mitigating the activity and impurities issues (occupational radiation exposure, environmental discharges, and material corrosion).

This general transverse knowledge of radiation protection stakes and industrial constraints allows us to justify the EPR™ design according to ALARA. Involvement of future plant operators may be necessary for specific licensing requirements and further improvements.

ALARA IN PRACTICE – REACTOR INTERNAL PARTS REPAIR AT THE TEMELÍN NPP

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Temelín NPP consists of two 1000 MW units, VVER type. During scheduled refueling outage 2GO11 in 2011, the need for inspection and reparation (grinding, polishing) of internal reactor parts surfaces, namely the reactor upper internals package, core barrel, and reactor vessel, occurred.

Based on comprehensive radiological survey, a decision was made to perform most of the work directly by the personnel. Several exposure reduction techniques were implemented applying the principles of time, distance and shielding. Usage of lead shielded hanging cage (gondola) and precise tuning of the water level in shafts, where the work was carried out, optimization of the trajectory of motion of the gondola, and careful work preparation together with time optimization lead to interesting ALARA achievements.

Very high dose rate at the reactor upper internals package did not allow any direct personnel intervention. A remote controlled grinder was therefore used as the only solution of the problem.

The paper describes the extent of radiation monitoring prior and during the work and its results, ALARA planning, and used personnel exposure reduction techniques. At last, personnel doses are compared with ALARA estimation.

OPERATIONAL EXPERIENCE OF THE REPLACEMENT OF PRESSURISER HEATERS DURING A FORCED AND A PLANNED REFUELLING OUTAGE

Matt LUNN, Guy RENN

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In early 2010 Sizewell B NPP (Westinghouse 4-Loop, 2nd Generation) experienced a small Loss of Coolant event, as a result of an emergent through wall leak from a Pressuriser Heater well, at the bottom of the Pressuriser Cell. As a consequence the plant entered an extended Forced Outage that replaced fifteen Pressuriser heaters that systematic inspection had showed were vulnerable to further in service leakage. A number of the Pressuriser heaters were physically damaged to the extent that their removal could not be effected by the normal process, from the bottom of the Pressuriser. Therefore a small number of heaters had to be removed from the inside of the Pressuriser; the engineering process developed required installation of an access platform inside the Pressuriser and more than 60 person-hours of occupancy inside the Pressuriser. The return to service safety case presented to the regulator required replacement of the remaining Pressuriser heaters in the following routine Refuelling Outage.

In the Autumn of 2011 the plant shut down for its eleventh Refuelling Outage. During the outage the remaining 65 Pressuriser heaters were replaced. The presentation will compare and contrast the radiological protection experiences of Pressuriser Heater replacement under emergent and planned outage conditions. In particular the presentation will review the radiation dose control and contamination control hazards, the control measures applied under the different conditions and the radiological protection outcomes. The presentation will describe the mock-ups and training used to familiarise crews and validate equipment and will describe specific radiation shielding, developed for the project.

PRIMARY SYSTEM DECONTAMINATION OF THE GERMAN PWR GRAFENRHEINFELD – PROCESS APPLICATION AND RECONTAMINATION EXPERIENCE

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Full-system decontamination (FSD) or replacement of steam generators and other large components in mature PWR plants results in large bare metal surfaces in the reactor system. The AREVA chemistry concept for long term dose rate reduction at such plants includes a passivation treatment for formation of high quality oxide films in the reactor system and high pH operation in combination with zinc injection during subsequent power operation. The effectiveness of these measures has been proven by plant operation experience. The systematic evaluation of plant-specific experience is used for further optimization of the concept.

These measures were also applied at the 1300 MW-class 4-loop PWR Grafenrheinfeld after a FSD with the AREVA decontamination process HP/CORD® UV in 28th outage. The passivation treatment consisted of zinc injection during plant heat-up and parallel adjustment of a high pH value. It was carried out in the plant operation condition “subcritical hot” for 200 hours and a pH(300°C) of 7.1. During the cycle, the pH value was increased to 7.4 and zinc injection was continued. This paper describes the characteristic features of the HP/CORD® UV process and its application at the PWR Grafenrheinfeld. The recontamination of the plant until the first refueling outage since the FSD is evaluated. The evaluation is based on routine chemistry and dose rate surveillance data and on the results of special measurement programs conducted at the plant before and after the FSD. Results of surface gamma activity measurements and fuel crud analyses are also described. Data is evaluated with regard to release, transport and deposition of corrosion products and nuclides in the reactor system. The plant-specific contamination sources and their pathways are outlined.

Compared to the first refueling outage of the plant in 1983, the dose rate level at the main loops were more than halved in spite of already activated core internals and crud on the fuel assemblies and control rods. The FSD resulted in total occupational exposure savings of approx. 4800 mSv in subsequent outages 2010 and 2011. Dose exposures for routine maintenance and inspection work, like non-destructive examination of steam generator tubing, decreased by a factor of 3.

SYSTEM DECONTAMINATION OF TWO BWRs

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In summer, 2011, a system decontamination of the reactor coolant and reactor cleaning system at Forsmark Reactors 3 was performed with very good results. The technique used was the CORD process. In spring 2012 the same technique was used when decontaminating the same systems at Forsmark Reactor 2. Even though both being BWR's, the two reactors are of different types. This presentation will describe the differences in decontamination result between two reactors with different water chemistry.

AN OVERVIEW OF ELECTRICITE DE FRANCE RESEARCH AND DEVELOPMENT PROJECTS IN THE FIELD OF OCCUPATIONAL RADIOLOGICAL PROTECTION

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The reduction of the largest individual doses (say greater than 10 mSv/year and concerning less than 1% of the work force including outside workers) and the reduction of the numbers of Personal Contamination Events are part of EDF Radiological Protection strategy for the next decade, in addition to overall radiological cleanliness improvement of its sites and collective dose decrease. To achieve these goals, tracking best practices worldwide and implementing them remain the fundamentals. However this is no longer sufficient as further progress becomes more and more difficult. Fortunately, even in a recession, techniques and scientific knowledge are still evolving, for example in the fields of low cost communication, advanced instrumentation or robotics. Applying these new means to the radiation protection of workers has been the core of EDF R&D projects in the recent and will remain so in the future. Besides, as human factors are the most often causes of RP adverse events, studies currently carried out in this field should lead to progress in eradication of RP events (often pinpointed by nuclear opponents), improving staff competencies, rules comprehension and organizational reliability.

In line with these considerations, the EDF presentation will address the main topics of its main three year R & D project EPURE. EPURE is devoted to contamination, dose reduction and addresses also RP training issues. It is also aimed at Remote Monitoring Systems development in order to support RMS implementation in the EDF fleet.

A special focus will be made on four items which illustrate the EDF approach:

- Development of advanced instrumentation featuring a handheld gamma camera and new friskers for workplaces with high background dose rates,
- Simulation of airborne contamination spread in a containment building in order to optimize the number and position of CAMs and detect as early as possible a contamination release,
- Remote monitoring systems developments, integrating teledosimetry, CAMs monitoring and control of engineering devices,
- Improvement of gammagraphic non destructive examinations safety through the analysis of human behavior and organizational reliability,
- Innovative developments for a realistic RP practical training as well as dedicated training programs including risk assessment basis and technical issues.

UNDERWATER DIVING REMOTE MONITORING IMPLEMENTATION

R. LEASURE (Braidwood NPP, USA)

2012 Ft. Lauderdale Distinguished Paper

This is a follow up to the paper presented on the KKL diving over exposure event that was presented at the Cambridge symposium. The paper will discuss the implementation of underwater telemetry that was implemented at KKL as a corrective action to the event. This paper will detail selection of equipment, classroom training and diving mock-up. It will also include key aspects of a successful diving program as well as several procedures from select US utilities.

LOWERING THE RADIOACTIVE AEROSOL EMISSION FROM FORSMARK 1, 2 AND 3

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The work done during 2009-2011 shows methods for air sampling in the exhaust ventilation system, measurements values and discuss radioactive aerosol sources together with possible actions to minimize the radioactive aerosol releases. Recommended actions can be either to prevent the spreading out or filtration. The work has been designed as a program with following studies/projects.

<u>Mapping of aerosol sources at Forsmark 1, 2 and 3</u>
A large number of air samples have been collected from different parts of the exhaust ducts in the reactor building, turbine building and the waste storage building. The air sampler equipment used is developed at Forsmark. The sampling is performed under isokinetic conditions where the air passes throw a particle filter which is analyzed with gamma spectrometric. The use of the portable air sampler and strategically sample points can also be useful in the search for aerosol leakage in the primary system.

<u>Improved status of exhaust air filtration at Forsmark 1, 2 and 3</u>
In order to prevent spreading of contamination it is essential that all pre-filter and HEPA-filter in the exhaust system works efficient. To achieve that Forsmark has increased the frequency of testing and changing the filters.

<u>Filtration of air during evacuation of empty fuel bottles</u>
Design of HEPA filter unit for filtrating during evacuation of empty fuel bottle. Analyze of air samples collected during evacuation indicates aerosol releases.

<u>Filtration of exhaust air during leakages from primary systems</u>
A leakage in the primary reactor systems can generate a large amount of radioactive aerosols. This study has investigated the possibilities and given recommendation for an enlarged filtration during aerosol leakages.

<u>Expanded use of mobile fans units with HEPA filter</u>
This study is an investigation about when and how to use mobile fan units in a nuclear plant and a market investigation for existing units.

<u>New equipment for cleaning the reactor pool</u>
A new cleaning machine for underwater cleaning of the reactor pool are under testing at Forsmark 3. So far the results show an improved and faster cleaning process.

<u>Filtrating of the exhaust air from the reactor hall</u>
A considerable part of the released activity during a year derive from the audit period. A study is therefore performed in order to evaluate the possibility of filtration the exhaust air from the reactor hall.

RESEARCH AND DEVELOPMENT OF GAS AND AEROSOLS DISPERSION IN THE RB AT EDF NUCLEAR POWER PLANTS

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Organizations, processes and materials used in EDF plants should enable to keep the dosimetry of workers below prescribed limits and values as reasonably achievable.

The collective risk of internal exposure was assessed and taken into account in designing facilities. For a real-time monitoring of atmospheric contamination in RB, EDF has set up a network of fixed and mobile Continuous Air Monitors (CAMs) which are disposed at workplaces and supervised outside the controlled area. The CAMs allow early detection of any atmospheric contamination and the alert of individuals in case of an unexpected increase in airborne radioactivity.

The efficiency and optimization of this device require knowledge of aerosols behaviour in ventilated areas. For 10 years, EDF has conducted fundamental studies and experiments to improve its knowledge of gases and aerosols dispersion in the RB.

To characterize the radiological conditions occurring in the RB, experiments were performed by EDF in collaboration with the IRSN laboratories in RB during outage. The radiological characterization was performed according to the ventilation state and the various work sites. The experimental measurements allowed a first validation of an aerosols transport and deposition numerical model which is implemented in the Code_Saturne ® CFD code, developed by EDF R&D with the collaboration of IRSN. This model was validated by comparing the transfer times and transfer coefficients of gas and aerosols predicted by the numerical model to the experimental results.

The CFD code is currently used to study specific situations. The aim of these studies is to improve the monitoring strategy in RB devices.

SUPERVISION OF PORTABLE VENTILATION UNIT (PVU): A WAY OF DEFENSE TO CONTROL THE HABITABILITY OF THE REACTOR BUILDING DURING OUTAGE IN EDF NUCLEAR POWER PLANTS

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Events of atmospheric dispersion of contamination have already happened in reactor buildings during maintenance operations of facilities wild outage in France and worldwide. These events may have a health impact on workers, are handle by evacuation of the reactor building until the radiological conditions are back to normal. These evacuations have a direct impact on the planning and organization of resources. Devices for monitoring air quality are installed to alert and evacuate personnel in case of detection of significant activity, but they can only act after the event and cannot avoid that it occurs.
The analysis of the origin of the main events encountered in the fleet of 58 French reactors showed that 90% of them are related to a failure in exploiting sites dynamic containment. These PVU defects can include a frank failure, abnormal power, filter damage.

To limit the occurrence of these events, the NPP of TRICASTIN developed in relation with industrials**, means of securing the PVU used in the EDF fleet. Among these means are the functions of automatic restart, alarm reporting and remote functional test. The integration of a communication interface also offers the possibility to connect the PVU to the main Radiobiological monitoring station of RB, being deployed on the EDF fleet. This monitor allows you to remotely view the functional state of PVUs and Radiological analyzers. It can also monitor the radioactivity of work sites volumes or surrounding PVUs. This radioactivity is due to the radioactive fines particles, iodine or gases contended into the collected air or accumulated on the filters.

The implementation of this device is a real deep way of defense, avoiding atmospheric contamination by detecting damage very early. The expected benefits are to improve health of workers, duration of outages and resources management.

** Our Partners for the project:
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RADIATION PROTECTION DURING DECOMMISSIONING OF NUCLEAR FACILITIES – EXPERIENCES AND CHALLENGES

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Decommissioning is the last phase of the life cycle of any nuclear facility. Today more than 450 facilities (nuclear power plants, research reactors) are finally shut down, under decommissioning or decommissioning has been already completed resulting in different technical and radiological end-states. As especially more and more nuclear power plants reach the end of their technical life time and thus will be finally shut down the number of decommissioning projects will increase in the next years. For example, of the 442 nuclear power reactors in the world in operation in 2006, 88 have been in operation for 30-40 years, 200 for 20-30 years, 109 for 10-20 years, and 45 for less than 10 years. Radiation protection of workers and the public – together with the management of radioactive waste and of spent fuel (if any is available) – is the central challenge during each decommissioning project. Depending on a manifold of influencing parameter, e.g. radiological inventory, complexity of the nuclear facility, decommissioning strategy or approach to structure the project, measure of radiation protection are different and specific for each individual decommissioning project.

Within this contribution an overview on experiences on radiation protection and best practice concluded from past decommissioning projects will be given and an outlook on future challenges in radiation protection during decommissioning will be provided. A special emphasis will be laid on the selection of decommissioning techniques, i.e. dismantling and decontamination technique, and on international activities to collection radiation protection experiences related.

RADIATION PROTECTION AT DECOMMISSIONING STAGE

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Decommissioning of NPPs including related radiation protection issues is of increasing importance in the near future. Due to the large activity inventory of the reactor pressure vessel (RPV) and the primary circuit of NPPs, radiation protection is of great concern with respect to decommissioning. Neutron flux calculations have to be performed to estimate the activation of the RPV material, the reactor internals as well as the residual contamination within the reactor and the primary circuit after a chemical decontamination. Specimen should be taken from the RPV wall and the RPV internals to validate the results of the calculations. Using these information, dismantling methods, radiation protection measures like shielding and the time schedule of work can be planned with special respect to the dose to personnel and the risk of contamination. Furthermore, the activity inventory serves as input for an optimized packing concept of casks and container. In this context AREVA NP has successfully performed decommissioning projects in the past. In this presentation respective examples will be given.

SHOVEL MONITOR FOR SORTING OF LOOSE DECOMMISSIONING MATERIALS FROM NPP

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To reduce the volume of contaminated loose materials to be monitored for clearance purposes, a pilot fast sorting monitor exploring the loader shovel was developed and tested within the NPPA1 decommissioning project. A pair of NaI(Tl) detectors and a measurement and navigation frame ensuring the loader shovel fixation in counting position is the basis of the developed fast sorting monitor. The navigation of loader and its shovel to the counting geometry should be as fast as possible. The monitored results shall be indicated by a prompt light indication system. MCNP 5 and ISOCS calculation tools were used for determination of ^{137}Cs full energy peak detection efficiency that is used for massic activity concentration evaluation. The estimated MDA for a pair of 2" x 2" Na(Tl) detectors and 30 s acquisition time is about 200 Bq/kg. However, due to the deviations in counting configuration from the supposed one (geometry, density, inhomogeneity of measured soil in the shovel) the uncertainty of measurements can be relatively high. Hence, the system is applicable for sorting monitoring, only.

Introduction

Relating to fulfill requirements to monitor large volumes of contaminated loose materials (mainly contaminated soil – CS) in short time but with sufficiently high sensitivity at activity concentration just above the free-release limits (300 Bq/kg for ^{137}Cs), a new monitoring system with high measurement productivity was required to be developed.

After considering several scintillation gamma spectrometry measurement arrangements, the loader's front shovel above a pair of NaI detectors was selected as the most suitable counting geometry for direct on site soil sorting monitoring at the site. A rigid support construction in the form of a "table", on which during the measurement a loader shovel with contaminated soil is placed was proposed and developed.

Purpose of the monitor is to sort out higher contaminated material (closely above release level) from the potentially releasable one. So in this way it helps to minimize volume of low level contaminated materials intended to clearance that compulsory requires free release measurement that is much more costly and time consuming (e.g. metrological certification of monitor). For clearance measurement the central drum monitor (3xHPGe detector) and soil belt monitor (2x LaBr) is used at the site. In such a way using of sorting monitor saves cost and time at contaminated soil clearance from the NPP site to the environment.

Goal of the paper is to describe the developed sorting monitor and discuss the experiences obtained during its trial testing.

Description of sorting monitor

The monitoring equipment further consists of a signaling box indicating the loader shovel position, process and results of measurement (Figure 1). The assembly is designed as transportable.

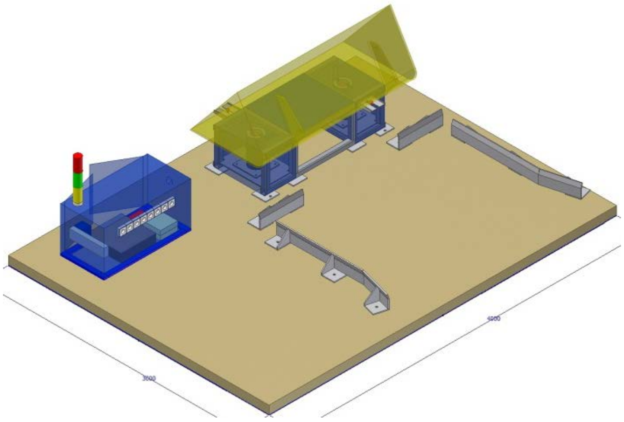

Figure 1 Design of monitoring assembly consisting of measuring table with elements ensuring correct loader shovel position and optical signalling module.

The correct loader shovel position during the measurement is ensured by mechanical navigation elements and by induction sensors of its position with corresponding optical signalization. The detection part consists of two shielded scintillation detectors 2” x 2” NaI(Tl) integrated into a protective support construction and adjoint to a 1024 channel gamma spectrometry system. Detection units in vertical position are placed into 5 cm thick lead cylindrical shielding. The shielding eliminates the interference radiation from the sides and from the undersurface (Figure 2). The shielding from above is ensured by self-absorption of measured material soil.

Next part of the assembly is a control PC with spectrometry and control SW. It serves for process control, acquired spectra gamma spectrometry evaluation and light signalling the measurement results including its saving to a hard disc. Monitoring consists of a determination of massic activity of ^{137}Cs [Bq/kg] and its comparison with set sorting levels, result of which is signalized by respective lights. Default set up sorting level is 500 Bq/kg of ^{137}Cs.

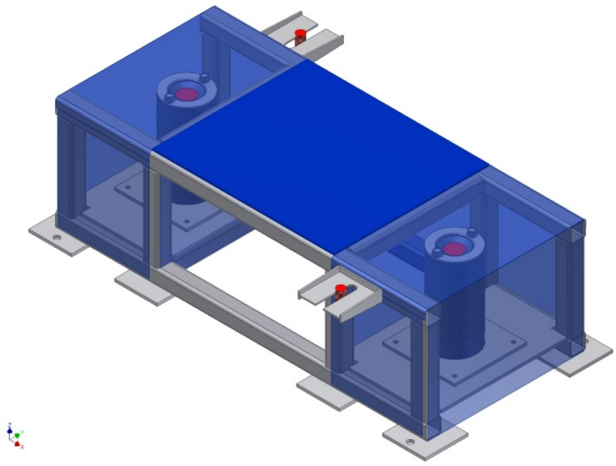

Figure 2 Positioning of shielded NaI(Tl) detectors and induction switches in the protective frame of the measuring table

It is assumed that the monitoring is performed with an existing loader, type UNC, with triangular profile and flat bottom of shovel, with dimensions about 75 x 45 x 180 cm (l x h x w). The shovel will be located in defined position to ensure measurements reproducibility. For detection efficiency calculation the cuboid shape with 30 cm thick soil layer was considered (Figure 3) for 661.6 keV (^{137}Cs).

Figure 3 Soil loaded into the UNC loader front shovel

The distance between the loader shovel's bottom and detector's head is about 5 cm. The detection sensitivity, 0.009 s^{-1} Bq/kg was estimated by MCNP and ISOCS calculations supposing the mentioned loader shovel effective counting geometry and homogenously contaminated soil parameters (soil density, shovel dimensions, etc.).

By on site tests was demonstrated that the proposed sorting monitors is sufficiently sensitive. MDA (minimum detectable activity according to Curie approach) was estimated at above mentioned simplified local conditions on the level 200 Bq/kg for ^{137}Cs at 30 s acquisition time. Depending on the operational conditions the monitoring capacity was estimated about 10 to 70 t per shift is expected.
Description of monitoring process

The measurement starts automatically after the loader shovel is put into correct counting position and stays in it. The correct position of the shovel is checked during the whole counting time. In the case that the shovel is removed from the counting position during the counting, the measurement is interrupted and erroneous loader shovel position is signaled.

After the end of acquisition time, the measurement is automatically evaluated and the measured value is indicated by colour signalling according to defined intervals. Particular acquired spectra and evaluation results of measurements are stored on a hard disc of the control PC.

The contaminated soil, sorted out as potentially free releasable into the environment, is transported to the free-release measurements working post, where the final free-release into the environment will be carried out.

Analysis of measurement uncertainty

Missing check of density, weight and thickness layer of the monitored materials in the shovel, as well as, short acquisition time results relative high measurement uncertainty. The uncertainty estimation was carried out on the basis of evaluation of the influence of differences between real counting geometry parameters and ones of calibration model by mean of ISOCS calibration tool.

The short acquisition time and a potentially inhomogeneous activity distribution in the volume of measured sample were considered, too.

Estimated value of standard combined uncertainty of measurement is about 45%. This value is acceptable for sorting monitoring purpose. Advantages of monitor are the high productivity, simple operation and low cost of monitor. Particular components of measurement uncertainty are introduced in the Table 1.

Table 1 Estimation of measurement uncertainty and its most important components

Source of uncertainty	Value of uncertainty
Net peak area uncertainty (at A_m = 300 Bq/kg ^{137}Cs)	10%
Uncertainty in consequence of the layer thickness change (assumed layer thickness is 30 ± 10 cm)	5%
Uncertainty in consequence of the density change (assumed density is 1.2 g/cm^3 with assumed change in range 0.8 - 2.0 g/cm^3)	7%
Uncertainty in consequence of layer shape change in the shovel (at partial pour out from shovel)	25%
Uncertainty of calibration method - ISOCS	6%
Uncertainty in consequence of non homogenously acidity distribution in monitored sample	34%
Standard combined uncertainty	**45%**

Results of trial operation of sorting monitor

Shovel sorting monitor ran in trial operate in 2011 with pair of NaI(Tl) 2” x 2”. Real monitor assembly is shown on the Figure 4 (with missing loader). Within this trial operation tree series of measurements with different contamination level of monitored soil in shovels were carried out. The measured ^{137}Cs data were compared with the laboratory gamma spectrometry analysis results for the soil samples taken out from particular shovels (Table 2).

Comparison of the shovel monitor results and the laboratory analysis results shows deviations in range up to 30 % that is in compliance with the expectation of uncertainty analysis. Comparison with a certified measuring instrument able to measure full volume of the shovel (e.g. drum monitor) is intended to do in next step.

On the basis of satisfactory results of the trial operation the shovel monitor is intended to innovate by replacement of used NaI(Tl) 2” x 2” detectors by a pair of 3” x 3” NaI(Tl) detectors. According to calculation similar to previous one it could ensures tree times higher sensitivity of measurement (S = 0.027 s^{-1}(Bq/kg)).

Table 2 Comparison of shovel monitor results of ^{137}Cs with gamma spectrometry analysis results of taken soil samples

Shovel monitor [Bq/kg]	Laboratory gamma spectrometry of samples [Bq/kg]	Deviation
128.0	113.4	13%
162.0	218.0	-26%
433.9	418.7	4%

Figure 4 Working place for sorting materials with shovel monitor

Conclusion

The loader's shovel fast sorting monitor was developed to increase the contaminated loose material monitoring and clearance capacity. The loader's front shovel monitor is based on two shielded NaI(Tl) 2” x 2” detectors, respective electronics and own control and Genie 2000 gamma spectrometry evaluation SWs. It includes respective positioning and light signalling system, as well.

Purpose of soil sorting by shovel monitor is to reduce the soil amounts to be monitored by clearance measurement at free release working post.

The carried out measurement uncertainty analysis and comparison measurements with HPGe laboratory gamma spectrometry of samples from shovel show that uncertainty of measurement is on the level of tens percents. The on-site monitor was demonstrated by a test trial as suitable for sorting purpose. Its advantage is short acquisition time (30 s), possibility to place it in close distance from the working place and so high monitoring productivity (tens of ton per shift) and in addition low cost of monitor.

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RADIOLOGICAL CHARACTERIZATION OF V1 NPP TECHNOLOGICAL SYSTEMS AND BUILDINGS - ACTIVATION

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V1 NPP at Jaslovske Bohunice site has been finally shutdown after 28 years of successful operation in 2006 (Unit 1) and 2008 (Unit 2). At present, both units are finally shutdown and since July 2011 under decommissioning license. The preparation of V1 NPP decommissioning has been supported and partly financed by the Bohunice International Decommissioning Support Fund (BIDSF), under the administration of the European Bank for Reconstruction and Development. From 06/2008 to 12/2011 AMEC Nuclear Slovakia, together with partners STM Power and EWN GmbH, carried out BIDSF B6.4 project - Decommissioning database development (DDB).

The main purpose of the B6.4 project was to develop a physical and radiological inventory database to support V1 NPP decommissioning process planning and performance. One of the specific deliverable tasks within the B6.4 project was deliverable D12 - Characterization of activated equipment and civil structures based on measurement, sampling and analyses performed on the samples. The scope of deliverable services within D12 task consisted of:

1. Categorization of activated components
2. Development of single working programs for their radiological monitoring and sampling
3. Preparation of sampling device and revision of all handling equipment
4. Dose rate monitoring and sampling of:
    - Civil structures from reactors shaft on both units
    - Components placed in HLW storage, (so called „Mogilnik“) - connection rods, absorbers of control rod assemblies and neutron flux measurement channels
    - Reactor pressure vessel and shielding assemblies at both units of V1 NPP, reactor internals from Unit 2 of V1 NPP
5. Analysis of samples
6. Determination of radiological inventory
7. Import of radiological data for activated components into DDB

During sampling, mainly remotely controlled sampling device and radiation resistant camera with LED lightening for visual checking of all performed activities was used. In total, 125 samples have been taken from all activated components. Subsequently gammaspectrometry and hard-to-detect analyses have been applied.

As a result, detailed radionuclide vectors for all activated components in accordance with requirements for near surface repository at Mochovce site have been determined. Based on radionuclide vectors, the radiological inventory including all radiological parameters for all activated components has been determined. Finally, obtained unique data on activated components, representing in total 1118 DDB items, have been imported into database developed within B6.4 project.

RADIATION AREA CLASSIFICATION AND SIZING OF THE STORAGE BUILDINGS OF USED-UP STEAM GENERATORS - PRACTICAL APPLICATION

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During Steam Generators Replacements, the used-up steam generators of each nuclear power unit are carried from the Reactor Building to the Building where they are to be stored. This storage building must be sized and positioned on nuclear sites so that dose rates, measured around it and near the site boundary, comply with the regulation on radiation area classification.

Studies have been carried out with the computer codes PANTHERE, MICROSKYSHINE and TRIPOLI in order to estimate dose rates on the setting up area of these storage buildings. These calculations take into account the decay of existing buildings and the influence of the future ones.

The results of these calculations have been integrated, after tests and analysis, in an Excel tool which offers a graphic display of dose rates around storage buildings. This user-friendly tool, accessible to people who are not specialized in radiological protection, allows justifying the sizing and the positioning of storage buildings by civil engineers with regard to the regulation on radiation area classification. It also enables to recommend an appropriate setting up configuration for future buildings in a real case.

Besides, a comparable tool has been developed to suggest an isodose curve during the transport of steam generators over the nuclear site (temporary radiological zoning). This curve, which is given to the plant operator the year before the outage, can be ripen just before the used-up Steam Generator removal from the Reactor Building thanks to a simple measuring range done on steam generators and filled in the Excel tool.

This method has the advantage of being transposable to other material and installation. Beyond the practical and user-friendly aspect of the made Excel file, this kind of studies can link radiological protection calculations domain (often considered as austere) to other technical professions of the nuclear domain (here, civil engineering) efficiently.

This study illustrates the role that radiological protection plays as support for other professions of the nuclear domain.

SOLIDIFICATION OF SPENT ION EXCHANGE RESINS INTO THE SIAL® MATRIX AT THE DUKOVANY NPP, CZECH REPUBLIC

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Peter Kopecký, CEZ a.s.

Based on the decision of the State Office for Nuclear Safety, the Dukovany NPP has been obliged to secure the efficient capacities for the disposal of spent ion exchange resins. Therefore, in September 2010, based on the contract with supplier company AMEC Nuclear Slovakia s.r.o. has begun with pumping and treatment of ion exchange resins from the storage tank 0TW30B02, situated at auxiliary building.
The SIAL® technology, developed in AMEC Nuclear Slovakia, has been used for the solidification purposes. This technology allows an in-situ treatment of various special radioactive waste streams (resins, sludge, sludge/resins, borates) at the room temperature. The SIAL® matrix and technology were licensed by the Czech State Office for Nuclear Safety in 2007.

On-site treatment and solidification of spent ion exchange resins at Dukovany NPP involves process of resin removal from tank using remotely operated manipulator, resin transportation, resin separation from free water, resin filling into 200 dm^3 drums and solidification into SIAL® matrix in 200 dm^3 drums using the FIZA S 200 equipment. The final product is observed for compressive strength, leachability, radionuclide composition, dose rate, solids and total weight. After meeting the requirements for final disposal and consolidation, the drums are being transported for the final disposal to the Repository at Dukovany site.

During the 3 month of trial operation in 2010, and the normal operation in 2011, the 123 tons of dewatered resins have been treated into 1200 drums, with total activity higher than 600 GBq.

Average weight of resins in the drum ranged from 89 – 106 kg and compressive strength limit (10 MPa) has already been achieved after 24 hours of fixation. The final measured strength values ranged from 19.0 - 34.7 MPa and real leachability values ranged from 0.03 – 0.65%, far below the 4% limit value.

Collective effective dose of all workers in 2011 was 12.6 mSv (6.2 mSv in 2010). Average individual effective dose in 2011 was 1.6 mSv (20 workers), and maximal individual effective dose was 3.0 mSv.

This approach allows fast, safe and cost effective immobilization and transformation of dangerous radioactive waste such as sludges and resins into the solid form, which is suitable for long term storage or disposal.

EPRI ALPHA GUIDELINE UPDATE

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In 2006, EPRI published the Alpha Monitoring Guidelines to standardize the industry's graded approach to monitoring for alpha contamination based on predetermined plant work environments. A minor update was published in 2009.

Since its original development, there have been several industry alpha events (e.g. Bruce Power, Oconee, Palo Verde) that have challenged the industry's alpha programs. Based on these experiences, the industry has requested EPRI to expand the scope of the Alpha Guideline to include not only monitoring and characterization requirements but to also include guidance related to proper work planning and analysis and gathering of environmental data subsequent to an alpha event to support accurate dose assessments. Additionally, the Alpha Guideline is being used more extensively by the international community. It has been requested that calculations included in the Guideline reflect updated ICRP dose parameters.

This presentation will discuss ongoing EPRI research and development activities to enhance worker protection related to alpha hazards. Recent plant experiences with alpha events will also be discussed to illustrate key lessons learned that will be incorporated into the guidance.

INTERNAL DOSIMETRY MONITORING: A COMPARATIVE STUDY OF THE EFFECTIVENESS OF BIOASSAY, NOSE-BLOW AND PERSONAL AIR SAMPLING MEASUREMENTS

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It is not possible to directly measure internal committed effective dose; therefore, indirect sampling programmes are required to infer a 'best estimate' of this dose. Nuvia dosimetry services have instigated a large-scale objective review of the results of bioassay (urine and faeces), nose blow and PAS programmes obtained over many years at the Harwell site. The objective of this review is to determine the relative effectiveness of these programmes in terms of their associated uncertainties and sensitivities in response to real exposure situations. The ultimate aim is to be able to provide guidance on how the use of these sampling programmes can be optimised for practical applications.

This study has been broken down into three separate phases; which includes a comparative review of all of the bulked routine PAS and bioassay data accumulated over about ten years; and a theoretical study of the expected uncertainties of the respective sampling programmes. The findings of both of these studies will be reported separately. This paper reports the findings from a review of how the sampling methods respond to known acute exposures.

A review of Harwell internal dose assessments has been carried out and cases with well-established intake dates selected. For these a comparison is made between the intake assessed from bioassay data, the PAS result on the day of the incident and nose blow data from the time of the incident. Correlations between intakes assessed from bioassay and PAS and intakes assessed from bioassay and nose blow are both found to be weak. Qualitatively, however, it is observed that when a high PAS activity is observed the occurrence of an intake is almost always confirmed by bioassay.

OCCUPATIONAL EXPOSURE IN A CANDU NPP: MANAGEMENT OF THE RISK FOR INTERNAL ALPHA CONTAMINATION

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The presence of transuranic (TRU) alpha-emitting radionuclides at nuclear power facilities can potentially impact many of the radiation protection aspects. These radionuclides, accumulated in some particular areas over time, can pose significant dose implications for work activities involving routine operations of ageing CANDU reactors.

Because TRU have high internal dose conversion factors for the inhalation pathway it is important to quantify their presence and assess their impact.

Detailed analyses of actinides in a CANDU plant revealed that the contamination levels of Am, Cm and Pu isotopes in the investigated locations are found to be in the following order with respect to their radioactivities: ^{241}Am > $^{239/240}Pu$ > ^{238}Pu > $^{243/244}Cm$ > ^{242}Cm. Each facility should characterize the transuranic radionuclide distribution and alpha activity levels in areas of the plant where alpha contamination may be present in order to determine which Area Action Levels (I, II, or III, as defined by EPRI) are initially assigned.

Contamination and airborne radioactivity surveys must be performed in order to evaluate the magnitude and extent of the radiological hazards in the workplace.

The assessment of TRU intake and associated doses is a very complex approach due to the difficulty in measuring TRU in whole body counting and bioassay samples, coupled with the increase in dose from inhaled TRU. Unlike beta/gamma-emitting nuclides, alpha-emitting nuclides are generally more difficult to measure, and often must be inferred from the presence of surrogate radionuclides. Scaling factors can be applied like surrogate relationships in order to establish a beta/gamma concentration value, above which an alpha analysis should be performed.

This paper provides information on the potential internal TRU contamination hazards at CNE Cernavoda and an assessment of the dosimetric consequences of an inhalation intake of the different TRU.

TRITIUM IN GERMAN LWRs – RELEVANCE ON RADIATION PROTECTION

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In Germany two different types of LWRs are in operation: BWRs and PWRs. With regard to the potential H-3 incorporation of staff there is a big difference between these two types of reactors due to H-3 production rate, ventilation rate, etc. While in BWRs H-3 incorporation normal is insignificant to effective dose, in PWRs there may be a not negligible impact on effective dose due to H-3 incorporation.

In this lesson I will show what are the parameters which determine the H-3 concentration in water and room air , what can we do to reduce the concentration in room air, how to calculate effective doses and I will show typical results of H-3 surveillance in German LWRs. Finally I will show our VGB-concept for surveillance of H-3 in German LWRs.

A STUDY ON THE SETTING UP A DOSE CONSTRAINT FOR RADIATION WORKERS IN KOREA NUCLEAR POWER PLANTS

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As required in ICRP 103 and IAEA BSS, the purpose of setting up a dose constrain for radiation workers in nuclear power plants of Korea in planned exposure situation is "to avoid severely inequitable outcomes of this optimization procedure, there should be restrictions on the doses or risks to individuals from a particular source. " A key issue on the setting up a dose constraint is how to define a particular source for nuclear power plants. There are also considerations of "(1) avoid severely inequitable exposure", "(2) key measures of optimization" based on the relevance intention of DCs set from ICRP/IAEA and "(3) minimization of changes for good practice (minimization of changes for existing exposure control, additional task and confusion of workers and managers)" based on the well-operated existing management system. ALARA committee and ALARA practical committee introduced DC per units of work (collective dose over 70 man-mSv) and that were analyzed for a very reasonable method which can be satisfied all items (1 ~ 3) above. In order to allow real-time monitoring, Korean NPP will set the DC and is going to operate improved radiation protection system after introduced DC on the statement of Korean nuclear safety policy in 2013.

KEYWORDS: dose constraint; ALARA; radiation worker; nuclear power plant

ALARA APPROACH DURING AREVA EPR DESIGN

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This presentation describes important aspects of radiation protection at the design steps of AREVA's new builds and highlights the ALARA methodology.

During basic design studies all radioactive sources corresponding to exposure situations are defined and several sets of values are postulated for different purposes: shielding design, radiological consequences of normal releases, and expected values closer to normal operational experience.

For the design of shielding walls, low target dose rates are used in frequently accessible zones of the controlled area: corridors, access rooms and accessible parts of the Reactor Building during power operation. Examples are shown for accessibility inside Reactor Building during power operation (service floor and the annular spaces) and for the handling operations above reactor pool during shutdown.

During detailed design, an introduction will be given to some other topics dealt with by AREVA for the new builds, such as the biological protections and corresponding equipment specification, the nuclide source term optimization (production and releases), the plant decommissioning and the estimation of occupational dose.

AREVA has a general transverse knowledge of radiation protection stakes and industrial constraints. This global approach allows us to justify the main EPR™ design choices according to ALARA. An involvement of future plant operators may be necessary for specific licensing requirements and further improvements.

NUMERICAL VALUE OF RADIATION PROTECTION PERFORMANCE INDICATOR OF NUCLEAR POWER PLANT - A PROPOSAL

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The radiation protection performance indicator (RPPI) as a numerical value can be established by using the definitions of dose constraints and collective dose budget for a Nuclear Power Plants or N-plats. The RPPI would be the best tool for comparative analysis of radiation protection program of a particular organization and with others. The RPPI formula is given as;

$$\text{RP Performance Indicator} = \frac{\text{Total Dose}}{\text{Dose Constraints / Dose Budget}} - 1$$

Total dose is collective dose for Station, collective dose of a group, individual dose and dose to the member of public of the critical group. The RPPI value shall fall between positive to negative based on the limit on the collective dose budget, dose constraints and the consumed total dose. The PI varies in range of ± 0.01 to ± 1 for collective dose of the N-plant, group and member of the public whereas 0 to ± 1 for the occupational workers. The PI shows % dose above or below the values set for exposure control. The formula is dependent on the dose constraints and collective dose budget for a particular time period which is first proposed by operator based on the operating experiences and reviewed/approved by the regulator. The formula does not address reactor power, operation & maintenance work and age management as these factors are being considered by operators during proposal. The proposed numerical RPPI would be more practical than qualitative comparison of collective dose because it involves efforts for optimization of radiation exposure by individual, group, Station and the regulator. The indicator will express the unequal distribution of individual exposure and enforce to operator for optimization. The proposed RPPI would be helpful for worldwide state members and in-house improvements for radiation protection. The RP performance indicator can also be categorized in extreme high, high, medium, low and extreme low for ± 0.10, ± 0.20, ± 0.30, ± 0.40 and ≥ ± 0.50 respectively for any N-plant or Station and to set future target for optimization of individual and public dose.

Key words: performance indicator, dose constraints, numerical, age management, regulator

REAL EXPERIENCE TRAINING™ FOR RADIOLOGICAL EMERGENCY RESPONSE

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A clear, current and demonstrable understanding of the correct operation of radiological dosimetry, survey, and spectrometry instruments is an obvious pre-requisite to their use, but there are inherent difficulties in training and verification of competency with such devices. The necessity to utilise radioactive sources and materials to gain experience of the response of real detectors and the ability to accurately communicate the readings obtained to decision makers can conflict with the requirement to minimise operator exposure. The need to instil confidence within local and national communities, Governments and stakeholders that emergency response procedures are well rehearsed, and that their safety concerns are adequately addressed, can be demonstrated in practice by the use of advanced simulation training systems in a manner that is otherwise impossible without an actual radiological release or sources .

This paper shall introduce new technological developments in the field of simulated radiological emergency and provide examples of best practice use including

- Types of radiation source/radioactive material simulation technology
    - Ultrasound
    - Magnetic
    - Fluorescent
    - Virtual
- Types of simulator
    - Contamination
    - Survey
    - Dosimetry
    - Spectrometry

New developments in simulator technology shall be described including

- The development of personal dosimeter simulators and the integration of their response with survey meter simulator readings
- Integration of simulators with teledosimetry systems
- Remote monitoring of simulator readings and control of simulated sources
- The introduction of integrated spectrometer simulators
- Wide area classroom (table top) and field exercises using simulation instruments

THE MONETARY VALUE OF THE MAN-MSV FOR KOREAN NPP RADIATION WORKERS ASSESSED BY THE RADIATION AVERSION FACTOR

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The monetary value of the man-mSv for operators of Korean nuclear power plants (NPPs) was calculated using a radiation aversion factor based on a survey of NPP workers. Initially, the life expectancy in the population is 79.4 y, the average age of cancer occurrence is 60 y, the average annual wage for an electric worker is 56,000 \$ y^{-1} and the nominal risk coefficient induced by radiation is $4.2E^{-5}$ mSv were used to evaluate the basic monetary value (abase) resulting in 45.6 \$ mSv^{-1}. To investigate the degree of radiation aversion, the subject of the investigation was selected as the working radiation workers in 10 NPPs in Korea (Kori 1–2, Yeonggwang 1–3, Ulchin 1–3 and Wolseong 1–2). In August 2010, with the cooperation of KHNP and partner companies, a total of 2,500 survey questionnaires to 10 NPPs (or 250 surveys to each NPP) were distributed to currently employed radiation workers. From these, 2157 responses were obtained between August and October 2010. The assessed radiation aversion factor and the monetary value of the man-mSv from the calculated radiation aversion factor were 1.26 and ~50 \$ in the 0–1 mSv range, 1.38 and ~200 \$ in the 1–5 mSv range, 1.52 and ~1,000 \$ in the 5–10 mSv range, 1.65 and ~4,000 \$ in the 10–20 mSv range and 1.74 and ~8,500 \$ >20 mSv.

KEYWORDS: monetary value; man-mSv; radiation worker; nuclear power plant

Introduction

As of 2011, 21 Korean nuclear power plants (NPPs) were in operation. Additionally, three units of OPR-1000 (optimised power reactor, 1000 MWe), the Korean standard NPP, and four units of APR-1400 (advanced power reactor, 1400 MWe), the next most common Korean nuclear reactor, are under construction.

The Korea Hydro and Nuclear Power Co., Ltd.(KHNP) has successfully carried out various design modifications, adoption of equipment and process improvements for the realisation of radiation protection optimisation in accordance with the As Low As Reasonably Achievable principle. Such efforts to reduce the radiation dose of radiation workers have resulted in the effective reduction of the collective dose of Korean NPPs' radiation workers from 1975 man-mSv/unit in 1985 to 641 man-mSv/unit in 2007. This exposure level is predicted to be maintained steadily at a low level(1).

However, there is no inherent monetary value of the man-mSv for operators of Korean NPPs that can quantitatively evaluate such efforts to realise radiation protection optimisation as reasonable and economical. International practices and cases are referenced on a case-by-case basis. Therefore, setting a unique monetary value of the man-mSv of KHNP is necessary as an essential means of decisionmaking in radiation protection optimisation.

The monetary value of the man-mSv, commonly referred to as the a-value, is widely used among NPP operators and state regulatory agencies as a secondary means of radiation protection optimisation realisation recommended by International Atomic Energy Association (IAEA) and International Commission on Radiological Protection (ICRP). The human capital methodology (HCM) and willingness to pay methods (WTP) are utilised to calculate the unit monetary value per man-mSv (\$/man-mSv) which is different for every regulatory agency and NPP operator as the factors and parameters used take into consideration the domestic socioeconomic conditions and unique characteristics of the NPP. In this study, the radiation aversion factor was analysed based on a survey of NPP workers and the socioeconomic conditions in Korea to evaluate the monetary value of the man-mSv that reflects the characteristics of NPPs in Korea. The monetary value according to the radiation exposure level is presented.

Materials and Methods

Survey objects and collecting information

To investigate the degree of radiation aversion, radiation workers in 10 Korean NPPs (Kori 1–2, Yeonggwang 1–3, Ulchin 1–3 and Wolseong 1–2) were selected as subjects. In August 2010, with the cooperation of KHNP and partner companies, a total of 2500 survey questionnaires were distributed to currently employed radiation workers at 10 NPPs (250 surveys to each NPP). 2157 responses were obtained between August and October 2010.

Survey contents and calculation of the radiation aversion factor

A minimum level of personal information regarding the survey subjects was investigated in order to examine the effects of age, years of employment and degree of awareness of the effect of radiation on the degree of radiation aversion. Also, the variation of aversion for each man-mSv level under equal annual salaries was investigated in the surveys which aimed to examine the consciousness and degree of radiation aversion.
After converting the degree of radiation aversion of the radiation workers into a scale of 1–5 for each man-mSv level, these values were then converted to a 1–2 scale and averaged to determine the radiation aversion coefficient (Figure 1, Table 1).

Risk aversion feeling		Original response		Recorded value
None	→	1	→	1.00
Very little	→	2	→	1.25
Some	→	3	→	1.50
Considerable	→	4	→	1.75
Complete	→	5	→	2.00

Figure 1. Calculation of radiation risk aversion value

Table 1. Aversion value scoring questionnaire by dose level

Dose level (mSv)	Aversion value (1-5)
1	( )
3	( )
8	( )
15	( )
>20	( )

The monetary value of the man-mSv is established with reference to the potential health effects depending on the level of exposure. The effect on an individual's health from cancer or genetic effects can be expressed as the loss in the individual's life expectancy. The economic loss to the individual that follows such health detriments are directly related to the monetary value of human life. HCM and WTP are applied in assessing the monetary value of the life expectancy loss. In this research, the monetary value of the man-mSv of NPP operators was evaluated utilizing HCM. The basic model equation presented in ICRP Publication 101 and IAEA SRS No. 21 are shown below [2, 3].

$\alpha_{ref}(d) = \alpha_{base}$ for $d<d_0$

$\alpha_{ref}(d) = \alpha_{base}(d / d_0)^a$ for $d \geq d_0$

Here, $\alpha_{ref}(d)$ is the monetary value of the man-mSv according to the individual exposure level d, α_{base} is the basic monetary value of the health detriment due to a unit dose, d_0 is the lowest limit of individual exposure with exposure aversion applied, d is the annual level of individual exposure, and a is the degree of exposure aversion or the risk aversion factor. Figure 2 presents the model equation. The y-axis shows what is reasonable to spend to prevent one man sievert, expressed in monetary units, and the x-axis shows the individual dose levels in mSv. [2]

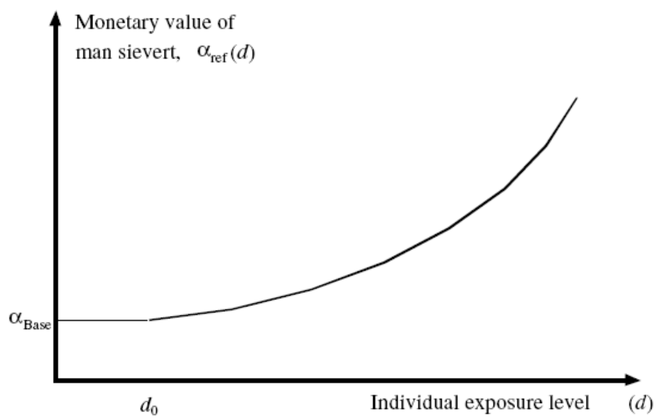

Figure 2. Model of monetary values of the man-mSv incorporating risk aversion and equity consideration

In August 2010, a survey was distributed to all radiation workers of 20 NPPs and 2,157 responses were collected. Table 2 shows the company distribution of the respondents.

Table 2. Distribution of respondents for the survey

Surveyed company	Profession	No. of respondents	Percentage (%)
KHNP	Operation	1,115	51.7
KEPCO KPS*	Machine/electricity-Inspection & maintenance	469	21.7
Radiation Contractor	Radiation management service	312	14.5
Samchang	Instrumentation-Inspection & maintenance	161	7.5
The rest		100	4.6

* KEPCO Plant Service & Engineering Co., LTD

In the case of occupational exposure, ICRP Pub. 101 suggests a risk aversion factor between 1.2 and 1.8 while the Nuclear Protection Evaluation Centre (CEPN) of France suggests a value between 1.2 and 1.75. As shown in Table 3, the risk aversion factor evaluated by CEPN was a=1.2 when the exposure level was 1-5 mSv/year, a=1.5 when 5-15 mSv/year, and a=1.75 when above 15 mSv/year.[4]

Table 3. Risk aversion factor "*a*" value by dose level (CEPN)

Dose level(mSv)	0-1	1-5	5-15	15-50
a	1	1.2	1.6	1.75

On the other hand, the Korea Institute of Nuclear Safety (KINS), a regulatory agency of Korea, announced the real monetary value of man-mSv of Korea in 2009 as shown in Table 4 below.[5]

Table 4. KINS's risk aversion factors and monetary values by radiation dose level.

Gross domestic product per capita (current price 2006) (US$)	18,374
GDP Deflator	111.7
Gross domestic product per capita (constant price)	16,449
Life expectancy in the population (years)	78.5
Loss of life expectancy induced by a radiation health effect (years)	18.5
Monetary value of one health effect	16,449 $\times$ 18.5 = 304,306.5 (US$)
Probability of occurrence of health effects associated with 1 Sv (Sv)	5.6 $\times$ $10^{\times 2}$
Risk aversion factor	
0 - 1.0 mSv	1.0
1.0 - 5.0 mSv	1.4
5.0 - 10 mSv	1.5
10 mSv and above	1.7
Monetary value of man-mSv exposure	(US$)
0 - 1.0 mSv	17,040
1.0 - 5.0 mSv	79,320
5.0 - 10 mSv	350,010
10 mSv and above	1701,620

Results and discussion

The value of α_{base}, the productivity loss due to life expectancy decrease, is shown in Table 5. In this research, the probabilistic risk factor of radiation was multiplied to the average annual wage of electric workers to reflect the characteristics of NPP radiation workers.[3] For the nominal risk coefficient of radiation, the value presented in ICRP 103 was used.

Table 5. Korean specific factors and basic monetary value (α_{base}) as of 2009

Life expectancy in population (A)	79.4 years(6)
Average age of cancer occurrence (B)	60.0 years(7)
Loss of life expectancy induced by radiation exposure (C = A-B)	19.4 years
Average annual wage for electric worker(W)	56,000$/year(8)
Nominal risk coefficient induced by radiation (P)	$4.2E^{-5}$mSv(9)
Basic Monetary Value (α_{base} = C×W×P)	45.6 $/mSv

As observable in Table 6, a majority (79.52%) of NPP workers in 2010 were exposed to less than 1 mSv. In the survey used in this study, a paired t-test was used to conduct a significance test on the differences in radiation exposure aversion feeling depending on the level of radiation exposure. The significance was determined at a significance level of 5%. The statistical analysis system (SAS ver 8.2) was used for the paired t-test statistical analysis.(10) 69.46% of the respondents were found to have had exposure of 1 mSv. It was found that there were differences in radiation exposure aversion feeling between respondents that were not exposed and those exposed to 3, 8, 15, and 20 mSv. On the other hand, there were no differences in radiation exposure aversion feeling for those with an exposure of 1 mSv and those with an exposure less than 1 mSv.

Table 6. Radiation dose distribution of NPP workers in 2010

	Dose range(mSv); *d*					
	$0 < d \leqq 1$	$1 < d \leqq 5$	$5 < d \leqq 10$	$10 < d \leqq 20$	$20 < d$	Total
Exposed people	9,957	1,720	565	258	22	12,522
%	79.52	13.74	4.51	2.06	0.18	100

In order to take into consideration the socio-psychological factors regarding radiation exposure, the public dose limit of 1 mSv/yr was set as the d_0 value. The radiation risk aversion factor *a*, which represents the degree of aversion against radiation exposure, was determined for different dose levels through the survey. The resulting α_{ref} values are organized in Table 7.

Table 7. KHNP's Risk aversion factors and monetary values by dose level

	Dose level (mSv)				
	0 - 1	1 - 5	5 - 10	10 - 20	> 20
Aversion factor	1.26	1.38	1.52	1.65	1.74
α_{ref} ($)	46	210	1,075	3,977	8,370

Conclusion

A comparison of the internationally and domestically managed monetary values of man-mSv reveals that most values used by NPP operators are 2-10 times greater than the values used by regulatory agencies. (3)

This finding is interpreted to be due to the regulatory agencies using the gross domestic product per capita and NPP operators using the average annual wage of employees in calculating the basic monetary value (α_{base}).
The risk aversion factors derived from the survey of NPP radiation workers are values based on the individual's radiation exposure and underlying perception of radiation. These risk aversion factors were used as an important basis in determining the monetary value of the by the NPP operators.
Additionally, while ICRP and KINS provide the risk aversion factor value of "1" in the 0-1 mSv range, the value investigated through this study resulted in “1.26”. The differences in the values represent a separate and additional aversion to radiation aside from the natural radiation dose even though this value is within the public dose limit.
The monetary value of the man-mSv is expected to contribute significantly in the NPP radiation protection optimization of KHNP. However, for convenient application in the NPP optimization of radiation protection, the values have been rounded up and down to provide a table of representative figures as shown in Table 8.

Table 8. Representative figures of monetary values of man-mSv by rounding up and down

	Dose level (mSv)				
	0 - 1	1 - 5	5 - 10	10 - 20	> 20
α_{ref} ($)	50	200	1,000	4,000	8,500

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ALARA CENTRE CONCEPT FOR KOREAN NPPS

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As a result of successfully carrying forward the dose reduction work about a change of design, Introduction of equipment, process improvement etc since 1980, It constantly goes down the personal average dose of Korea reactor worker and group dose per unit which are each from 5.1 mSv, 1,975 man-mSv/unit in 1985 to 1.1 mSv, 641 man-mSv/unit in 2007. And it's the best level in worldwide. However, (it is the situation) we need to develop "The ALARA system" of a new tool to continue with dose reduction. The existing ALARA activities and methods were hard to get the best result of ALARA because this structure is organized separately such as each plants or company, so we couldn't share with information about Best Practice. Also, computerize of ARARA Process is surely necessary to performance ICRP Publication 103(dose constraint and monetary values by dose level). Therefore, in 2012, we will strive to make continuously best examples about dose reduction of the reactor workers in highest level through developing "ALARA center" software and performing it which can benchmark best practice of domestic and foreign country and based on ICRP Publication 103.

KEYWORDS: ALARA centre; radiation worker; nuclear power plant

DEVELOPMENTS IN THE DELIVERY OF THE LEGAL ELECTRONIC DOSIMETRY SYSTEM IN EDF ENERGY

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The Thermofisher Electronic Personal Dosemeter (EPD) has been the Approved Legal dosimeter system for EDF Energy (ex British Energy) since 1998. This paper reviews the operational history and developments of the EPD since its introduction within EDF Energy. It describes the recent developments (2011) in the EPD system which were driven by computing hardware and software developments and identifies lessons learned in the deployment and upgrade of the computing infrastructure of the EPD system.

STUDY AND DEVELOPMENT OF DETECTION DEVICES FOR DIRECT CONTAMINATION MEASUREMENTS IN A HIGH GAMMA BACKGROUND AT EDF NUCLEAR POWER PLANTS

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Since more than one year, EDF and CEA started a new collaborating development. This new development consists in the design of a beta contamination monitor for surface and body contamination measurements. This monitor includes a gamma background rejection and is intended to be used in several places inside nuclear power plants.

The EDF requirements concern performances improvements of the equipment currently used for the surface contamination detection. This equipment is operated especially for contamination control of these facilities, both for equipment or personnel, on sites during operations in nuclear controlled areas. Meanly, measurement campaigns take place during the outage and inside the reactor building (RB), where the gamma background could be significant and with a high level of variation. The background level can vary from few micro-Sievert per hour to several hundred micro-Sievert per hour.

Depending on the location of the work site, the surface contamination detection limit is 4 Bq.cm2 down to 0.4 Bq.cm2. The current equipment used available may not reach these detection limits by direct measurement, in presence of a significant gamma background. The body checks are often performed from the Reactor Building and surface contamination checks are carried out by indirect measures with smears to ensure a low level gamma background.

The main EDF objectives are to have a detector available to achieve these measurements as close as possible to work areas, where the level of the gamma background is up to 30µSv/h.

The collaboration between CEA and EDF focuses on the development of a complete subsystem including the detector of the monitor associated with a real time gamma background rejection.

The first step consists in validating the principle of detection, evaluating the metrological and first laboratory performances.

The second step will strength on optimizing the system in industrial and to make several tests on site to qualify the product in real situations.

The first Lab performances and description of this new contamination monitor will be presented in this paper.

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IMPROVEMENTS IN NEUTRON DOSIMETRY AT NUCLEAR POWER PLANTS

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The most widely adopted techniques world-wide for occupational neutron personal dosimetry are:

- Solid State Nuclear Track detectors using polycarbonate (CR-39) to detect proton recoils (for intermediate and fast neutrons) or alpha products (from $^{10}B(n,\alpha)^{7}Li$ reactions in a Boron rich irradiator, for thermal neutrons)
- Albedo dosemeters which incorporate a thermal neutron sensitive detectors (typically ^{6}LiF thermoluminescence detectors (TLD) or more recently using optically stimulated luminescence (OSL) detectors (aluminium oxide coated with $^{6}Li_2CO_3$)) which measure the reflected (albedo) thermal neutrons backscattered from the body.

Both techniques have advantages and disadvantages in practical situations which have been discussed widely and are summarised in this paper.

Landauer has been providing personal neutron dosimetry services to clients operating nuclear power plants for a number of years. Recently we have worked with a number of our clients to improve the reliability and accuracy of personal neutron dosimetry measurements. This has included:

- Assessment of the extent of doses to workers in nuclear power plants (typically 99.9% receive less than 1 mSv).
- Improvements to the dosemeters and their analysis.
- Comparison with other types of dosemeters and manufacturers.
- Characterisation of environments in which neutron exposure is of concern.
- Optimisation of the type of neutron dosemeter to be used and instruments for monitoring the environment.
- Field calibration of dosemeters.

This paper will outline this work and draw together lessons learned and recommendations for the optimal assessment of personal dose in nuclear power plant environments.

ABSOLUT-3: AN AREVA TOOL FOR THE ANTICIPATION OF TRITIUM MANAGEMENT AT PWRs

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Tritium is produced in large amounts in nuclear power plants. In PWRs, the tritium concentration in the primary coolant is directly related to the energy supplied by the reactor. Among many possible reactions, tritium results from ternary fission in the fuel and also from neutron activation reactions with lithium and boron isotopes dissolved in the primary coolant as well as with naturally-occurring deuterium in the primary coolant.

Most of the tritium produced in the fuel rods is usually retained within the fuel and is not released into the environment at the reactor site. It is released during fuel reprocessing, if that practice is carried out. As currently there is no industrial method for tritium trapping and due to the low radiotoxicity of tritium, tritium produced in the primary coolant is usually entirely released, without treatment, in the effluent streams according to the waste management practices and discharge authorizations at the plant. On the one hand present liquid releases of tritium should be decreased for public and radiological acceptance; on the other hand new fuel managements and power levels of new-builds and up-rates lead to higher boron concentrations, and thus higher tritium production rates. But due to possible storage problems of tritiated products and contamination of groundwater Safety Authorities are unwilling to grant any further increase in tritium releases.

OUTAGE ACTIVITY TRANSPORT MONITORING AT CERNAVODA NPP

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

Dose rate and gamma spectra surveys were performed for the first time at the reactor faces, vertical feeder and moderator heat exchanger of Cernavoda Unit 1 during Outage in May 2010, 19 days after reactor shutdown.

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

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

The intensity of the radiation fields at the reactor faces of Cernavoda Unit 1 is similar to that at Darlington units; the radionuclides distributions are, however, unique.

Similar determinations were performed at Cernavoda Unit2, during planned outage in May 2011.

SOURCE TERM MONITORING PLAN USING CZT IN KOREAN NPPs

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Recent advanced CZT (Cd-Zn-Te) spectral analysis and measurements which reduce dose in order to monitoring the source term inside circuit by real-time spectrum analysis are rising technology by France EDF that achieved outstanding dose reduction, and lately many other countries including the United States are embraced that technology. The CZT technology is able to provides quite useful information for improving the process such as shut-down chemistry, zinc injection and the purification of contaminated power plants units and the advanced radiological information about contaminated area, working area, and radioactive materials in real time, and it make possible to step forward of radiation protection activities. CZT devices can get the best results when the spectrum measuring techniques, the geometry analysis technique, and the prediction methods for exposure dose are combined organizationally and each steps are automated. The Radiation health research institute, therefore, is planning to conduct a source term modeling through the CZT measurements and geometry analysis after identifying the variation of source term characteristics for Korean NPPs from 2012.

KEYWORDS: source term reduction; CZT; nuclear power plant

MODELING OF CORROSION PRODUCTS CONTAMINATION IN PWR WITH THE CALCULATION CODE OSCAR V1.2

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During shutdown operations and maintenance of PWR, personnel exposure is mainly caused by activated corrosion products, which make the major contribution to dose rates in the vicinity of systems and components. The reduction of collective dose is one of the main objectives for nuclear industries. The reduction of contamination and the understanding of mechanisms leading to contamination are therefore of prime importance. For about 30 years, a French collaboration between CEA, AREVA and EDF has been carried out in order to develop OSCAR (Outil de Simulation de la ContAmination en Réacteur), a calculation code for the modeling of corrosion products, fission products and actinides behavior in nuclear reactor circuits. This code integrates the more recent R&D results based on theoretical and experimental results in test loops and is calibrated on in-situ deposits measurements performed in PWR systems. This tool allows to predict the contamination of PWR and is really useful for design and operation procedure improvements for the optimization of the existing units as much as for the new-built units. The version 1.2 of this code is the current version and the most accomplished version available nowadays.

This paper focuses on corrosion products contamination. It presents the main mechanisms implemented in the OSCAR calculation code as well some results of calculation compared to measurements. The impact on corrosion products contamination of specific key parameters (operating and design parameters) is also presented.

EPRI RADIATION FIELD AND SOURCE TERM REDUCTION INITIATIVES

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The global nuclear industry continues to strive to reduce radiation exposure to nuclear plant workers. One effective way to reduce worker exposure is to minimize the radiation field hazard itself through source term reduction. Source term reduction technologies must balance the complex, multi-dimensional chemical process of corrosion product generation, release, deposition, activation, transport and incorporation against fuel design and plant optimization. The effectiveness of source term reduction technologies can be impacted by materials mitigation strategies, materials of construction, core design, power uprating, cycle extensions, and changing chemistry regimes. A full and fundamental understanding of how current and future chemical control and purification methodologies affect the corrosion product life cycle is needed to continue making gains in reducing radiation fields.

Radiation field reduction will be reached through a balanced approach of optimized chemistry control, fuel design, and materials, and it must continue to evolve supporting plant life cycles of 60 years and beyond. Technologies necessarily must mitigate the generation of radiation fields at each stage of the life cycle from generation of parent nuclides (corrosion and release) to deposition of the activated corrosion product in out-of-core surfaces. This presentation will discuss ongoing EPRI research and development activities to understand the generation of radiation fields and the technologies available to reduce radiation fields.

DECONTAMINATION CONCEPTS FOR DECOMMISSIONING

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Decontamination prior to decommissioning and dismantling is worldwide the most accepted approach and here especially the Full System Decontamination (FSD) during the post operation phase. FSD in the context for PWRs is defined as the chemical decontamination of the primary cooling circuit, in conjunction with the main auxiliary systems.

FSD provides the following advantages for the decommissioning:
- minimization of radioactive inventory in an early stage
- minimization of personnel dose exposure during decommissioning planning and performance
- minimization of radiological sampling for dismantling planning for rad-waste and storage
- minimization of nuclide vector numbers
- increase flexibility on decommissioning strategy
- support dismantling techniques (e.g. for RPV)
- reduce waste for post treatment and increase 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 result in higher reliability of scheduling and budgeting

AREVA has long-term experience with Full System Decontamination for return to service of operating nuclear power plants as well as for decommissioning after shutdown. Since 1976, AREVA has performed over 500 decontamination applications and from 1986 on, decontaminations prior to decommissioning projects which comprise virtually all nuclear power plant (NPP) designs and plant conditions:
- NPP designs: HPWR, PWR, and BWR by AREVA, Westinghouse, ABB and GE
- decontaminations performed shortly after final shutdown or several years later, and even after re-opening safe enclosure
- high alpha inventory and or low gamma/alpha ratio
- main coolant chemistry (e.g. with and without Zn injection during operation)

Fifteen decontaminations prior to decommissioning projects have been performed successfully to date. The lessons learned of each project were consequently implemented for the next project.

This paper will outline the comprehensive approach for decontamination for decommissioning (DCD), lesson learned and highlights of previous FSDs performed prior to decommissioning, with respect to:
- planning scenario
- application window
- decontamination area
- waste considerations
- positive results for subsequent decommissioning and dismantling activities

WASTE MANAGEMENT OF INTERNAL PARTS (VIDEO)

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A video that presents the waste management chain of internal parts in Forsmark. This video shows how the internal parts are cut to pieces in the reactor basin, almost without any water contamination at all, then packed in a cassette. The cassette is then moved from the reactor basin and placed in a so called BFA storage tank and then vacuum dried. The BFA tank is then placed in a moving vehicle which moves the tank to its long time storage position.

UKRAINIAN NPPs' EXPOSURE DOSES MANAGEMENT

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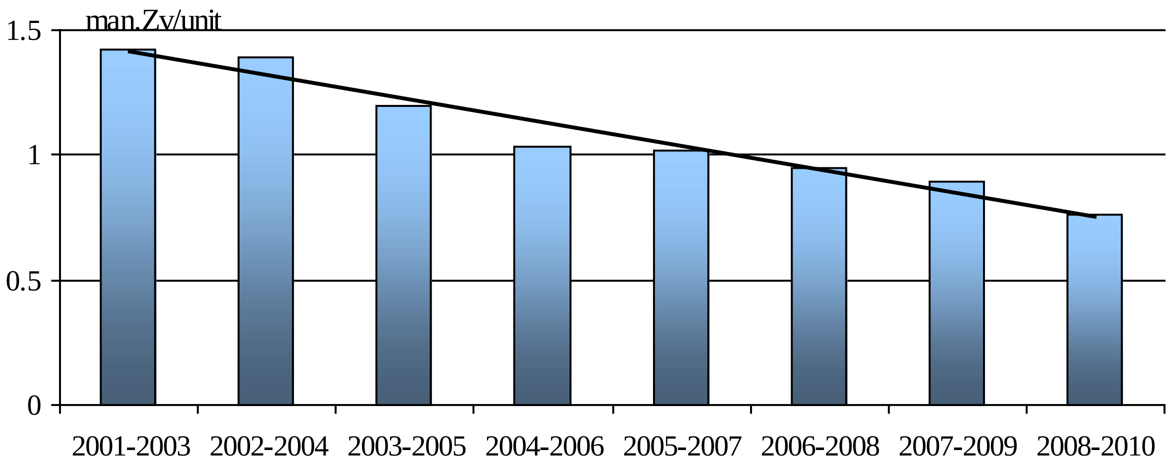
The analysis on dynamics of collective exposure levels shows steady trend towards decrease over recent years.
The chart below shows the dynamics of average collective annual exposure doses per one unit type PWR (three year averages) over the last ten years.



Principal component of a captured personnel exposure dose is a dose received during hazardous radiation works associated with maintenances of units. Management of these doses are critical in determining personnel exposure reduction efforts on NPPs.
At the beginning of each calendar year, radiation safety specialists on each NPP perform analytical dose assessment works and determine quotas for collective exposure doses per each unit and total for a nuclear power plant. Quotas are determined with respect to previous data obtained during maintenance works on NPPs. Also, these quotas take into account the anticipated volumes of hazardous radiation works on the unit during an upcoming calendar year.
Numerical quota values are finalized by a NPP managerial stuff and registered in a corresponding executive paper.
The requirement to comply with predetermined quotas of collective doses proved to be an efficient instrument, which stimulates development and further implementation of organization and technical measures directed on exposure levels decrease.
Quota approach for collective exposure levels reduction allows executing basic principles of radiation safety, which are limitation, optimization, and justification.

INNOVATIVE PLASTIC SCINTILLATOR DETECTOR FOR CONTAMINATION MONITOR UNDER A GAMMA BACKGROUND

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The design, preparation and first test of a new contamination detector are presented. The goal of this project is to be able to measure with a high level of accuracy beta contamination as low as 0.4 up to 4 Bq/cm² in a high and fluctuating gamma background level of approximately 30 µSv/h. The measurement should be performed within 2 to 3 sec with precision of less than 60 %, with a detection surface of approximately 200 cm². β/γ discrimination is performed via a dual plastic phoswich scintillator. This detector setup has been totally reconsidered and does not display any dead zone between the two scintillators. Fully ΔE/E discrimination is therefore possible and the access to the total beta energy is available.

ΔE/E discrimination is based on the coupling of a rapid decay time and thin scintillator with a slow decay time thick scintillator. The making of this phoswhich addresses two issues: the fluorescent molecules properties on one hand and the search of the best matrix coupling which affords good scintillation properties without any dead zone on the other hand. Thus, several couples of fluorescent molecules were tested so as to extract two couples of fast (decay time < 4 ns) and slow (decay time < 80 ns) provided by adapted fluorophores. The emission wavelength for both couples was centered on 420 ns but could be extended towards 500 nm. The matrix was the key of the project and had to display both excellent scintillation properties (by conversion of the energy of the nuclide to fluorophores) and insolubility for one layer to another during the preparation process. Another challenge was to refine the fast scintillator to be the best homogeneous thickness as possible along the whole surface.

Light collection, coating of the surfaces and aging of the scintillators under various constraints (low and high temperatures, UV, irradiation) will also be discussed. Preliminary lab performances of this new phoswhich will be presented hereby.

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RADIATION DOSIMETRY RELATED TO SPENT FUEL CASKS HANDLING AND STORAGE ACTIVITIES AT TEMELIN NPP

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In 2010 dry spent fuel storage at Temelin NPP was commissioned. The design capacity of the storage is 152 spent fuel storage casks. So far five CASTOR© 1000/19 casks have been placed in the storage.

The cask handling procedure will be shortly described; related radiation exposure of personnel and results of individual casks mandatory survey measurements will be presented.

Neutron-photon mixed field was characterized in terms of ambient dose equivalent in the vicinity of the stored casks by portable instruments. Additionally, neutron spectra were measured by means of a Bonner sphere spectrometer, and used to derive the corresponding neutron ambient dose equivalent. The photon and neutron personal dose equivalents rates were evaluated by albedo and electronic dosimeters, the results will be compared and discussed in relation to results obtained by survey meters and neutron spectra measurements.

COMPONENTS REMOVAL FROM FUEL STORAGE POOL IN REACTOR BUILDING: PREPARATORY ACTIVITY TO THE VESSEL DISMANTLING OF CAORSO NUCLEAR POWER PLANT

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Caorso Nuclear Power Plant (NPP), a 870 MW_e Boiling Water Reactor (BWR) belongs to the second generation of NPP and it was built in early seventies.
In 1990, the decision of permanently shutting down its activities was made. Since that date, the safety of the plant and its systems has been guaranteed, in order to protect the population and the environment. Nowadays, Caorso NPP is in decommissioning.
One of the main tasks currently performed in the facility is components (waste) removal from fuel storage pool in Reactor Building (RB). The actions to be implemented are:

- components removal;
- components cutting and volume reduction;
- components packaging in special transport and storage cask.

The activated/contaminated components involved in this removal are:

- Local Power Range Monitor system – hot section;
- Steam dryer lifting lug;
- Steel nuts, bolts and clips;
- vibration sensors;
- Irradiated fuel channels;
- Control rods;
- Feedwater sparger

Total activity referred to all components is 1.46E+16 Bq, in particular 1.04E+16 Bq of Co-60.
The optimization of individual and collective dose to the workers is designed through ALARA philosophy.
The VISIPLAN 3D ALARA planning tool is developed and designed by SCK•CEN (Belgian Nuclear Research Center) as a dose assessment tool, enabling the user to calculate the dose in a 3D environment for work scenarios. This software is very successful in the ALARA field.
The method used for dose assessment is based on a point-kernel calculation with an infinite media build-up correction. The tool calculates:

- the effective dose in a specific point;
- the dose account for different work scenarios, defined by the ALARA analyst taking into account the worker's position, work duration and subsequent geometry and source distribution changes in a 3D computer simulation of the work place;
- iso-dose curves in a defined work area (multi-sources and different shielding materials).

The VISIPLAN 4.0 tool proves to be useful in the dismantling design phase.
Applying the ALARA principle, we have defined:

- work planning;
- work duration;
- shielding, if necessary;
- maximum individual dose;
- collective dose.

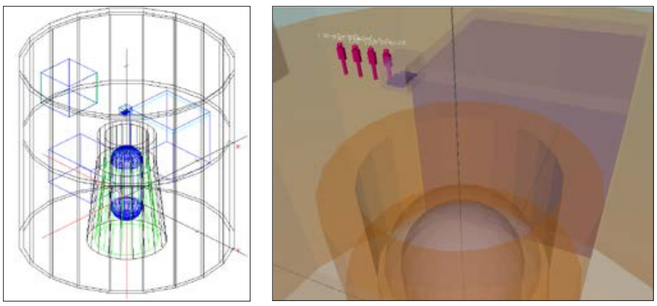

Figure 1 – Work Area 3D-Model

HARMONIZATION OF USE AND CALIBRATION OF RADIATION MEASUREMENT EQUIPMENT IN THE SPANISH NUCLEAR POWER PLANTS

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Radiation Protection of workers department (APRT) of the Nuclear Safety Council (CSN), in the framework of the joint group of Radiation Protection (composed by Spanish nuclear representatives (UNESA) and the CSN), entrusted a study about radiation measurement equipment to radiological control of people and materials in the way out of controlled zone and the double fence used in the Spanish nuclear power plants (SNPP). The object of this study was the harmonization of use and calibration of the aforementioned equipment.

UNESA carried out two different tasks. On the one hand, several aspects related to radiation measurement were harmonized in the SNPP. On the other hand, UNESA prepared comparative tables showing information of all SNPP about radiation measurement equipment, where they are placed, amount, model and manufacturer, physical detection principle, calibration type (alpha, beta or gamma), isotope, geometry, detection efficiency, calibration periodicity and use and calibration procedures.

The CSN analyzed the study presented by UNESA. As a result of this analysis questions to solve came up and, especially additional tasks in order to reach measurement common criteria for all SNPP.

The main task of the CSN is to analyze the study of UNESA in order to harmonize and unify criteria in all SNPP about contamination control, equipment and reinforcement of measurement equipment resources and the harmonization of the alarm set-points.

CONSIDERATION OF THE RADIATION PROTECTION IN ENGINEERING ACTIVITIES AT CIPN (ELECTRICITE DE FRANCE)

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The organization has led EDF to entrust CIPN(*) with the backfitting of the French standardized plants. These modifications are designed to CIPN, with the support of industry partners, before being deployed to the different reactors, taking into account the technical characteristics of the reactors (bearings 900MW, 1300MW) and an optimized deployment planning. The modifications represent 10% of the collective dose Park EDF into play by the operations, maintenance or modifications.

Increased modification programs, capacity requirements in the field of radiation protection have led to give since the 1990s an ever more important for Radiation Protection in organizations and activities of engineering, especially modifications.

This resulted in:

- Establishing a process to deal with the Radiation Protection as well as other technical activities and describes the implementation of ALARA in the design and deployment of changes, for the completion of the modification and operation Later reactor
- The constitution of a specialized team CIPN implementing simulation tools for calculating, measuring means and methods to guide design decisions and ensure consideration of risk and optimize interventions,
- The establishment of contractual requirements requiring industrial partners to take account of the Radiation Protection in the design and implementation of activities assigned to them.

With examples of modifications, it illustrates the inclusion of the Radiation Protection CIPN, the difficulties that require attention in view of a modification program just as important.

(*) CIPN: Engineering center of nuclear operating

CONCEPT AND APPLICATION OF RECORDING LEVEL FOR MONITORING OF INDIVIDUAL DOSES AT NPPs

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ICRP and IAEA recommend using of recording level (RL) for individual monitoring of occupational doses – the level, above which recording of the doses should be required (doses below the RL are to be accepted equal to zero).

According to data presented in the UNSCEAR reports "Sources and effects of ionizing radiation", now the concept of RL is applied all over the world. However, an ambiguous wording of this ICRP and IAEA recommendation led to the fact that the values of the RL, the approaches to their establishment, order and purpose of their application differ from country to country.

This situation doesn't affect reliability of the basic dose limits monitoring; however, it can lead to non-correct interpretation of monitoring results during optimization of radiation protection or comparative analysis of exposure indicators of various NPPs, units or departments.

In this report evolution of key points of RL conception over the past 30 years was analyzed, estimation of influence of RL on collective dose calculations was done and recommendations for using of RL at NPPs were offered.

OCCUPATIONAL DOSE ESTIMATE APPLIED FOR NEW BUILDS – AREVA APPROACH FOR THE REACTOR BUILDING

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The aim of this paper is to communicate on AREVA's methodology for occupational dose estimate at the design stage of the new builds.
The designer is positioned at the centre of the optimization initiative in order to justify the EPR™ reactor as an improved reactor design according to ALARA and in relation to the best units currently operating.

After a presentation of the other existing approaches, the AREVA method proposed for the EPR™ dose prediction analysis for the Reactor Building will be detailed.
The calculation of the overall occupational dose is performed by an analytical methodology which consists in totalizing the contributions of each elementary task foreseen in the EPR™ outage schedule.
This method requires a highly detailed database of the tasks expected to be performed during operation of the plant. Such database allows observing the impact of a modification of one input parameter involved in calculation on the results of collective dose.

This methodology provides an alternative option to the others methods which focus on the optimization of high priority activities.

IMPLEMENTATION OF NEUTRON SHIELDS FOR ACCESSIBILITY INTO THE EPR™ CONTAINMENT DURING POWER OPERATION – EXAMPLE OF THE PRIMARY PIPING NEUTRON PROTECTIONS

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EPR™ reactor design allows for accessibility within the containment of the Reactor Building during power operation. The economic asset of such access permits to perform preparation maintenance activities during power operation, reducing in consequence the duration of outage.

A "two-room" concept has been foreseen to contain potential airborne contamination from the primary equipment bunkers to the accessible areas, i.e. annular spaces and service floor. Moreover thick concrete walls and floors ensure the mitigation of the two main sources during power operation: neutrons and nitrogen-16 gamma rays.

Nevertheless the eight openings in the reactor pit for primary piping allow for neutron propagation from the reactor pit to the primary bunkers. Therefore neutron shields are implemented around the eight primary pipes at the exit of the reactor pit to reduce as much as possible the neutron dose rates within the primary bunkers and thus within accessible areas.

A neutron shielding material resistant to high temperatures in the neighborhood of the primary piping had to be foreseen for this neutron shields. Moreover removability of the eight shields was foreseen to ensure that the reactor pit openings are accessible for periodic inspections on the welds between the reactor pressure vessel and the 8 main coolant lines. Consequently a dedicated design considered the ventilation of the RPV and the interfaces with surrounding lay-out in the primary bunkers, particularly the primary piping insulation and connected nozzles.

USE OF DOSE CONSTRAINTS AT CHASMA NPGS

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In pursuant to regulatory requirement, Chashma Nuclear Power Generating Station (CNPGS) has established Dose Constraint Values (DCVs) to ensure proper control on radioactive effluent releases and that public doses attributed by these releases are ALARA. This paper provides the method used to establish DCVs and analytical model used to calculate Annual Discharge Limits (ADLs). Official DCV is 0.26 mSv has been set for each unit of CNPGS. However, even a lower DCV that is 0.20 mSv is considered for calculation of ADLs. The ADLs on radioactivity of tritium and beta gamma emitters for liquid releases have been calculated and verified on the basis of DCVs of 0.08 mSv. Calculation of ADLs based on DCV of 0.12 mSv for gaseous releases containing particulates, iodine & noble gases has been completed and is at final stage of verification.

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