Radiation Protection

OECD

AIEA IAEA

# Occupational Exposure Management at Nuclear Power Plants

OECD Nuclear Energy Agency
International Atomic Energy Agency

ISOE

INFORMATION SYSTEM ON OCCUPATIONAL EXPOSURE

## Fourth ISOE European Symposium
## Lyon, France
## 24-26 March 2004

AEN
NEA

NUCLEAR • ENERGY • AGENCY

Radioactive Waste Management

# Occupational Exposure Management at Nuclear Power Plants

## Fourth ISOE European Workshop
## Lyon, France
## 24-26 March 2004

Organised by the European Commission and
the European Technical Centre (CEPN)


NEA No. 6069

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

**ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT**

The OECD is a unique forum where the governments of 30 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

The OECD member countries are: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Korea, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Slovak Republic, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation's statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

* * *

*This work is published on the responsibility of the Secretary-General of the OECD. The opinions expressed and arguments employed herein do not necessarily reflect the official views of the Organisation or of the governments of its member countries.*

**NUCLEAR ENERGY AGENCY**

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full member. NEA membership today consists of 28 OECD member countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, the Slovak Republic, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

## FOREWORD

The Information System on Occupational Exposure (ISOE) was created in 1992 to provide a forum for radiation protection experts from both operating organisations and national regulatory authorities to discuss, promote and co-ordinate international co-operative undertakings in the area of worker protection at nuclear power plants. The ISOE System is promoted and sponsored by the OECD Nuclear Energy Agency (NEA) and the International Atomic Energy Agency (IAEA), which provide a Joint Secretariat for the programme.

Since 1997, ISOE has developed a programme of annual workshops and symposia for radiation protection professionals from all types of nuclear power plants. Attendees also include contractors and regulatory staff. The workshops and symposia are held alternatively in North America and in Europe. The European workshops are co-organised by the European Technical Centre and the European Commission, which provides a substantial financial contribution. The IAEA supports the workshops and symposia by providing financial help for participants from countries participating in ISOE through the IAEA and also for participants from target countries of two IAEA Technical Co-operation Projects aimed at enhancing occupational radiation protection in nuclear power plants.

These workshops and symposia have given hundreds of professionals an opportunity to listen to oral presentations (about 30 in each workshop), exchange information, share ideas and learn from others. The workshops' concept, with contributions from and for the radiation protection professionals, has proven to be very effective. The discussions on selected topics in small groups in Europe and the practical ALARA training sessions in North America have contributed to the success of the programme.

The 2004 International ALARA Symposium was held from the 24-26 March 2004 in Lyon, France. The symposium with the theme "Occupational Exposure Management in Nuclear Power Plants" was organised by the European Technical Centre in order to provide a global forum to promote the exchange of ideas and management approaches to maintaining occupational radiation exposures "as low as reasonably achievable" (ALARA). The symposium was sponsored by the European Technical Centre (ETC), the OECD/NEA and the IAEA. The workshop enjoyed several varied oral and poster presentations.

The success of this Workshop is largely due to the important organisational support from the Électricité de France.

# TABLE OF CONTENTS

# SESSION 1

# INTRODUCTION OF JNES IN JAPAN

**K. Komori**
Japan Nuclear Energy Safety Organization (JNES), Japan

## Introduction

The Japan Nuclear Energy Safety Organization (JNES), an incorporated administrative agency, was established on 1 October 2003, as a technical support organisation to the nuclear regulatory authority, the Nuclear and Industrial Safety Agency (NISA), with the mission to ensure the safety of nuclear installations for energy use.

JNES's activities include inspection of nuclear installations, safety analysis and evaluation, emergency preparedness support, technical survey, tests and research for ensuring nuclear safety, and information analysis, evaluation and transmission. The studies on new approaches to ensuring nuclear safety on the basis of the latest technological knowledge are also within JNES scope of responsibilities.

## Outline of the Organization

### *Major tasks*

1. Inspection of nuclear installations:
   - pre-service inspection, periodical inspection, fuel inspection, periodical safety management examination and safety management inspection for welding;
   - pre-service inspection, periodical facility inspection, inspection of waste disposal facilities, welding inspection, probation of waste disposal, probation of nuclear material transportation.
2. safety analysis and evaluation of nuclear installations;
3. support for nuclear emergency prevention and response;
4. survey, tests, and research for assuring nuclear safety;
5. analysis, evaluation and transmission of nuclear safety related information.

## Organization chart

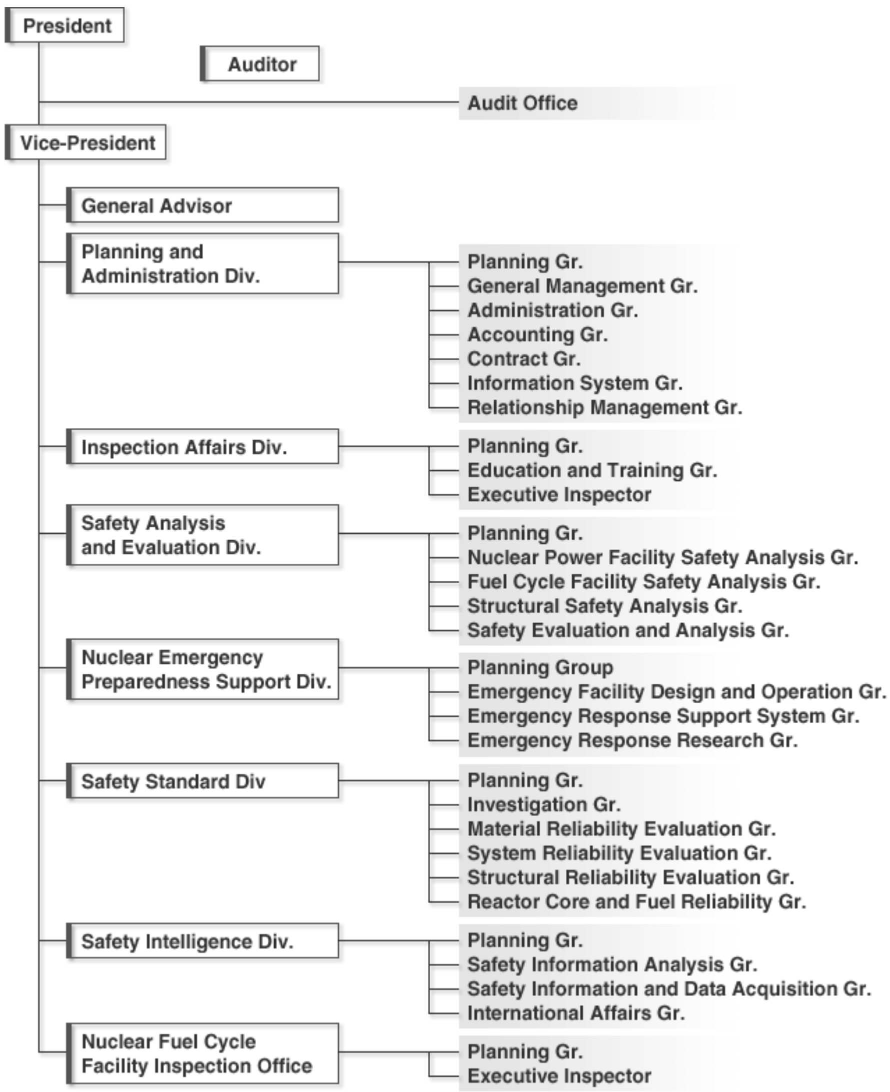
### *The number of management and staff*

Board of Managers is composed of six senior managers; one President, three Vice-Presidents and two Auditors.

The staff counts about 420 (as of 1 March 2004): **Budget scale:** 12.2 billion yen (Funds granted from the government for FY2003).

Note: The budget for FY2003 is a half of a regular fiscal year since JNES was established in October 2003, in the mid the FY.

**Organization chart**



## Role and mission

The role and mission of our Organization is, in brief, “to ensure the public safety against the potential hazard accompanying the use of nuclear energy through efforts of expert engineer groups”.

JNES, just established in October 2003, is to accomplish its role and mission as an organisation which is the inheritor of the fruit of the human wisdom accumulated over the long history. In these days in Japan, the public has some anxiety about nuclear safety and strong interest in the safety ensuring measures has become common. In this context, JNES was established for improvement of nuclear safety regulatory system in securing the foundation for ensuring safety.

JNES, expert organisation in nuclear safety, will faithfully accomplish its mission in order to meet the expectation of the public in cooperation with the nuclear safety authority, the Nuclear and Industrial Safety Agency, Ministry of Economy, Trade and Industry (NISA/METI).

## Major tasks

### *Inspection activities*

Inspection activities are actions which objectively confirm the compliance of the licensees with the nuclear safety regulation. They play an essential role in ensuring the nuclear safety. JNES conducts nuclear installation inspections of various kinds, periodical review of safety management and probation. JNES is required to perform these inspections in fair and strict manners with flexibility.

In the framework of recurrence preventive measures against illicit acts discovered in August 2002 regarding the falsification of self-imposed inspection results at nuclear power plants, the Electric Utilities Industry Law has been amended to provide for self-imposed inspections by licensees as legal obligation and to introduce the periodical audit of the safety management system. The latter ensures the neutrality of the judgments on the results of now compulsory self-imposed inspection, on one hand, and forms a framework for improving the self-imposed inspections, on the other. JNES is assigned to conduct these inspections and audits.

Along with the proper and reliable conduct of the inspections and audits which JNES took over from the Government or its designated agencies, JNES will also establish a reliable audit system for quality assurance and develop objective and reasonable audit criteria in order to effectively implement the recently introduced audit of the safety management system and to enhance its effectiveness. JNES expects that all the efforts described above contribute to enhancing the nuclear safety and to obtaining the public confidence.

### *Activities concerning analysis and evaluation of the safety of nuclear installations*

For the government’s safety examination to grant licenses to nuclear installations, JNES conducts safety analyses evaluation of the adequacy of design to ensure the safety of the installations even with abnormal transients and accidental events. In doing this evaluation, it is necessary to verify, by independent analyses, the adequacy of nuclear licensees’ safety analyses. The independent analyses are performed with different analysis codes (computer programs) than those used by the nuclear licensees.

In addition to the appropriate independent analyses performed for the safety examination conducted by the government, JNES continuously develops and maintains the analysis codes and evaluation methods required for the future independent analyses on the installations, which are expected to be soon subject to the safety examination.

Besides the safety examination for license granting, JNES independently evaluates the safety analyses such as periodical safety reviews, probabilistic safety assessments (PSA) and the accident management. Some of these analyses are performed by the licensees for the purpose of safety improvement; various events actually occurred in the real plants are analysed and evaluated to verify the safety of the nuclear installations.

***Supporting activities for the nuclear disaster preparedness***

In case of accidents at nuclear installations, the government, local communities and nuclear licensees must respond promptly and properly in a concerted manner to minimise the effects of the accidents on the public and environment. For this purpose, it is important to conduct drills on a regular basis to confirm the emergency procedures to be used by the relevant parties in an emergency, and to prepare and properly maintain the installations and equipment necessary for an emergency situation.

JNES develops plans for nuclear emergency preparedness drills and performs the work such as coordination among the related organisations in cooperation with the Nuclear and Industrial Safety Agency. JNES also prepares emergency response facilities (off-site centers) in the vicinity of the recently established reactor facilities, maintains and manages them properly to get always available the off-site center and the Emergency Response Support System (ERSS), and establishes a system for response against accidents. In addition, JNES performs the studies and researches on the nuclear emergency preparedness and provides the staff of associated organisations with training and education.

***Study, test and research to ensure the nuclear safety***

Because reactor installations are highly complex system, the knowledge and information associated to ensuring their safety derive from wide scope raging from design, operation to decommissioning. For this reason, it is necessary to regularly collect the latest knowledge and compile the data and information as the basis for the safety regulations for the purpose of proper implement-tation of nuclear safety regulations. In addition, for regulations based on scientific and rational judgment, it is indispensable to adequately organise the knowledge into standards or rules, to reflect them in the review of the regulatory system and to improve review criteria.

JNES conducts the studies, tests and researches necessary to achieve the above goal. These activities will be conducted with clear vision as to their outputs to be used for the standards or rules to be established.

To put it concretely, the activity scope covers both domestic and international standards and rules in such fields as reliability evaluation of facilities, aging countermeasures, seismic reliability of facilities, fuel characteristics, safety of nuclear fuel cycle facilities, safety of decommissioning, safety management of radioactive waste disposal, transportation of radioactive materials, human factors, accident management and publicly invited proposals of studies and researches.

In addition, for the conduct of tests and researches, an evaluation system by a third party is introduced to check for selection of the proper theme, development of an activity plan, control of the progress and objective evaluation of the results. When it becomes clear that accomplishment or useful application is difficult to expect from ongoing programme, such programme will be immediately re-examined and, if a decision is taken to terminate it, the programme will be aborted promptly. Thus, JNES will always remain conscious of the administrative needs of the safety regulation.

### *Collection, arrangement and provision of information to ensure the safety*

It is important to make effective use of information on failures or events occurred at the similar facilities in order to ensure the safety of reactor facilities. It has been recently found effective for the prevention of more significant event to carefully gather and analyse the operational information on minor incidents that cannot be called failures or events. Therefore, it has become necessary to extract useful information to be applied to other facilities by accumulating and analysing not only failures and events, but also operational information on minor incidents less significant than events at both the domestic and foreign reactor facilities.

JNES is planning to accumulate information on ensuring of safety through international network, to establish databases for ensuring of safety, to analyse the information, to study actions to be taken to ensure safety and to make useful proposals.

Finally, since it has become an important issue to increase the transparency in safety regulatory administration in order to restore the public confidence in ensuring nuclear safety, JNES will provide easy-to-understand information regarding the safety regulation.

## Closing

Since activities in the nuclear safety area have become further specialised and gained international dimension, which reflects the latest technical progress, JNES, recognising itself as a professional organisation, is keenly aware of its extremely large and important role in this area. With its foremost concern being to assure the public safety, JNES will respect transparency and give plain explanations based on scientific and rational judgments. We feel that the public expects from nuclear experts to show the true picture of the nuclear technology, and believe that serious discussions and plain explanations by nuclear experts will surely gain the public confidence. Upon recognition of these facts, our staff will fulfill their jobs with all their strength to meet the public's expectations for ensuring nuclear safety.

# OPERATIONAL RADIOLOGICAL PROTECTION AND ASPECTS OF OPTIMISATION

**E. Lazo**
OECD Nuclear Energy Agency, France

**C. G. Lindvall**
Barsebäck Kraft AB, Sweden

**Abstract**

Since 1992, the Nuclear Energy Agency (NEA), along with the International Atomic Energy Agency (IAEA), has sponsored the Information System on Occupational Exposure (ISOE). ISOE collects and analyses occupational exposure data and experience from over 400 nuclear power plants around the world and is a forum for radiological protection experts from both nuclear power plants and regulatory authorities to share lessons learned and best practices in the management of worker radiation exposures. In connection to the ongoing work of the International Commission on Radiological Protection (ICRP) to develop new recommendations, the ISOE programme has been interested in how the new recommendations would affect operational radiological protection application at nuclear power plants. Bearing in mind that the ICRP is developing, in addition to new general recommendations, a new recommendation specifically on optimisation, the ISOE programme created a working group to study the operational aspects of optimisation, and to identify the key factors in optimisation that could usefully be reflected in ICRP recommendations. In addition, the Group identified areas where further ICRP clarification and guidance would be of assistance to practitioners, both at the plant and the regulatory authority.

The specific objective of this ISOE work was to provide operational radiological protection input, based on practical experience, to the development of new ICRP recommendations, particularly in the area of optimisation. This will help assure that new recommendations will best serve the needs of those implementing radiation protection standards, for the public and for workers, at both national and international levels.

This paper will provide the practitioner's perspective for the implementation of an effective programme of optimisation of worker radiation exposures.

Operational radiological protection focuses very strongly on assuring that exposures to workers and the public are maintained "as low as reasonably achievable" (ALARA). While this concept is central to the day-to-day management of exposures, the complex nature of exposures and exposure situations mandates a flexible approach to the implementation of radiological protection actions. The increasing participation of various stakeholder groups in decision-making processes further suggests

the need for flexibility to assure the appropriate incorporation of these views. Although philosophy, policy, regulations and guides are necessary as a framework for operational applications, these guiding tools should remain rather non-prescriptive to allow the radiological protection practitioner to appropriately find the optimum option for radiological protection on a case-by-case basis.

In this context, radiological protection professionals are very interested in the current development of new recommendations from the International Commission on Radiological Protection, ICRP. To assist in this development, the NEA/IAEA Information System on Occupational Exposure (ISOE) developed, through its Working Group on Operational Radiological Protection (WGOR) this report. The objective of this work is to remind the international radiological protection community, and the ICRP, of the practical aspects or radiological protection that should be reinforced by any new ICRP recommendations, and to identify areas where further practical guidance would be useful. Several key messages, that are elaborated in the body of the report and supported by practical examples in the report's annexes, have been developed.

In the area of public exposures, it is clear that the objective of radiological protection profess-sionals is to use a process of optimisation to protect members of the public, workers and the environment. Minimisation of dose is not the objective. The ALARA philosophy and the use of best available technology (BAT) are both used in optimising exposures. Within the process of optimisation, it should be remembered that protection options that decrease public exposure at the expense of significant worker exposures are not seen to be ALARA. Collective dose is an effective planning tool for comparing options, but, particularly with respect to public exposures, is not used to assess public detriment.

Worker exposures are also managed using a process of optimisation. Workers themselves contribute significantly to work planning, using their operational experience to improve work efficiency. Worker collective dose is an extremely useful tool for worker exposure management. To effectively manage doses, flexibility is needed for controlling collective dose and for assuring that individuals are equally protected. As such, having an individual dose limit/constraint of 20 mSv/a can be restrictive and can actually lead to increases in collective dose. A key aspect to worker exposure management is the effective empowerment of the workforce. This can result in several positive effects that are closely linked together, including; lower doses, higher safety, higher efficiency, lower costs, and more efficient use of resources. While it should be remembered that national and plant-specific approaches to the implementation of work management practices may differ significantly (responsebility, distribution of tasks, etc.), the objectives of work management can be achieved by many approaches. Work management will include the consideration of many aspects of worker health and safety than simply radiological protection.

The optimisation process, as applied to both public and worker exposures, is inherently judgmental and case-by-case, using quantitative and qualitative approaches. As such, flexibility in guidance for the application of optimisation is needed. Optimisation of dose, below a given dose constraint, focuses on the process, not on the results. As such, the site-specific philosophy for the implementation of optimisation and ALARA may be equivalent while yielding different results. It would be very useful to have guidance on the types of criteria that should be considered when judging the effectiveness of an ALARA/optimisation programme.

These things being said, however, the application of a generic level, on the order of a few 10s of μSv/a, below which the need for regulatory control, if any, would be reduced, would be welcomed by the nuclear industry. It should be noted, however, that, particularly as these levels would be applied in decommissioning operations, any levels that are eventually chosen for clearance levels, and regulatory requirements for release measurements for verification of compliance with these criteria should not result in excessive worker exposures. Worker exposures should be key elements that are considered when national decommissioning policy is developed.

Finally, the nature of international recommendations implies a certain level of agreement on common approaches. To assure that common approaches leave sufficient national and local flexibility, the level of common approaches and understanding needed to effectively optimise public and worker doses needs to be discussed. One area where the need for guidance is clear is the national and international management of itinerant worker exposures. Here, it is understood that the responsibility for the management and optimisation of worker doses lies at all levels:

- The management and optimisation of worker doses is the responsibility of the worker's employer.
- However, the facility causing worker exposure is responsible for optimising all doses received at that facility.
- National regulatory authorities are responsible for monitoring worker doses and their compliance with dose limits.

This being said, expanding the use of practical tools, such as "dose passports", should be explored nationally and internationally.

In any case, the ISOE programme encourages the open dialogue of the broad radiological protection community on the development of new international recommendations. Because of the broad impact that such recommendations could have on national radiological protection regulations and implementation, it is suggested that any new ICRP recommendations should be reviewed from the legal standpoint, which will probably be necessary at the country level, and for their practical implications before they are finalised.

# REGULATORY REQUIREMENTS FOR RADIATION SAFETY IN THE DESIGN OF A NEW FINNISH NPP

**K. Alm-Lytz, O. Vilkamo**
Radiation and Nuclear Safety Authority (STUK), Finland

## Introduction

There are two operating nuclear power plants in Finland, two BWR units at Olkiluoto site and two PWR units at Loviisa site. These reactors were commissioned between 1977 and 1981. The total electricity capacity in Finland is about 15 GW. In 2003, nuclear power plants generated one fourth of Finland's electricity. Despite of the diversity of the electricity generation methods, Finland is highly dependent on imported energy. Electricity consumption is estimated to increase and the demand for extra capacity has been estimated at about 2 500-3 000 MW by 2010 [1]. It should also be taken into account that a considerable proportion of the production capacity constructed in the 1970s must be replaced with production capacity of new power plants in the near future. In practice, the climate politics commitments made by Finland exclude coal power. Therefore, the capacity can be increased significantly only by natural gas, nuclear power and biofuels [1].

## Licensing a new nuclear power plant in Finland

The licensing process of a new nuclear power plant in Finland is shown in Figure 1. The project of the fifth Finnish nuclear power reactor was formally started in May 1998 with Environmental Impact Assessment (EIA) process. The EIA process was completed in January 2000. Results of the EIA were used to support the application for a Decision in Principle, which the electricity generating company TVO submitted to the Ministry of Trade and Industry in November 2000. The Finnish Government made in January 2002 a Decision in Principle, which concluded that constructing of a new nuclear power plant unit in Finland is in line with the overall good of the society. The Finnish Parliament ratified the decision in May 2002. Based on this decision, TVO was authorised to continue preparations for the construction of a new nuclear power plant unit.

In October 2003, TVO decided the plant site to be Olkiluoto and in December 2003 TVO made a contract with a consortium of Framatome ANP and Siemens to build a French-German reactor concept EPR (European Pressurised Water Reactor). TVO submitted the application for Construction License to the Ministry of Trade and Industry in the beginning of 2004. The Construction License evaluation process takes approximately one year, and the construction works on-site could start at the earliest at the beginning of 2005. Based on TVO's schedule, estimated construction time is about four years. The

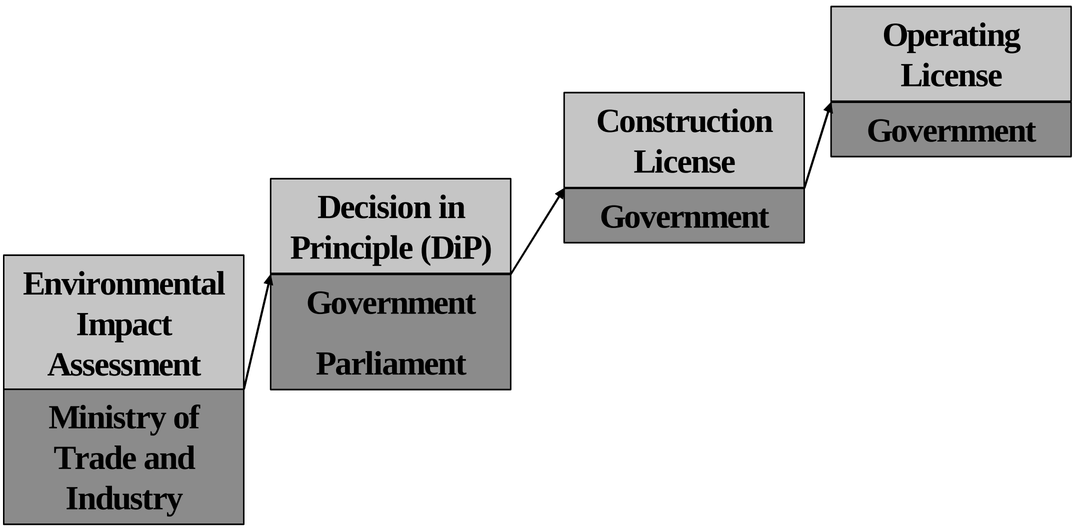
Operating License evaluation process takes approximately one year, and thus, the new unit could be in operation in 2009 if no unexpected delays occur.

At the same time the application for Construction License was sent to the Ministry of Trade and Industry, TVO submitted so called licensing documentation to STUK. According to the Finnish Nuclear Energy Decree Section 35, these documents include:

- preliminary Safety Analysis Report (PSAR);
- proposal for a classification document;
- description of quality assurance during the construction;
- plans for physical protection and emergency preparedness;
- plan for safeguards;
- description of the applicant's arrangements for the regulatory review by STUK;
- other reports that STUK considers necessary.

Based on the review of these documents, STUK prepares its statement on safety and safety assessment, which will be submitted to the Ministry of Trade and Industry. STUK's positive statement on safety is a prerequisite for the Government to grant the Construction License.

**Figure 1. The licensing process of a new nuclear power plant in Finland**

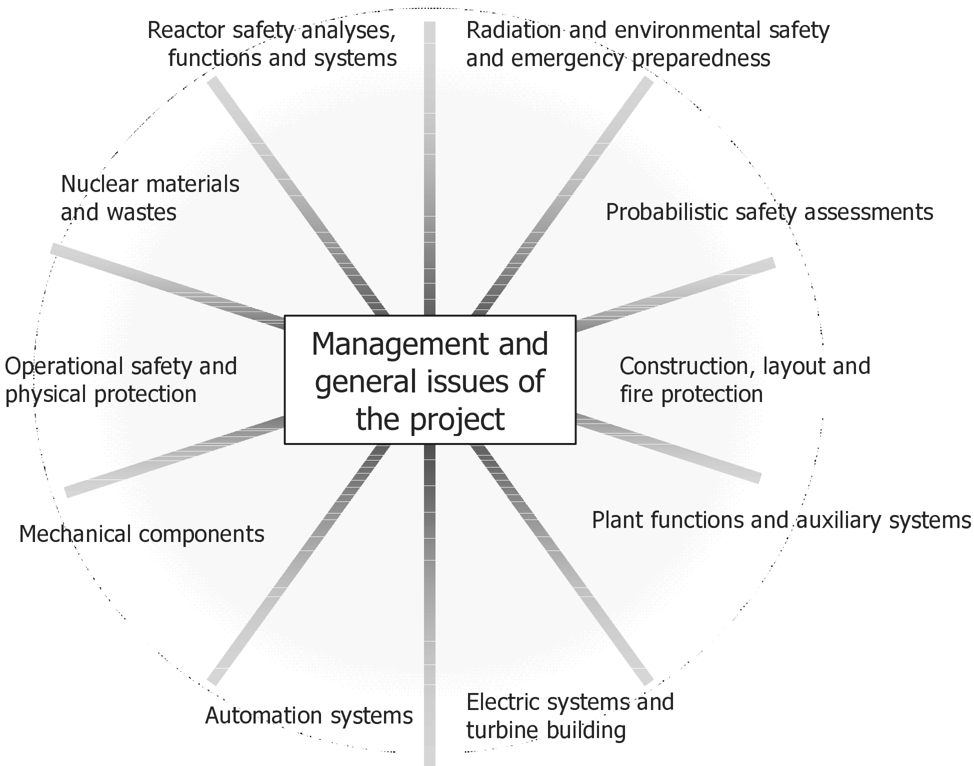


**FIN5 project at STUK**

After the decision in principle (DIP), the Radiation and Nuclear Safety Authority (STUK) established a project group to co-ordinate the license application regulatory review process of the fifth Finnish NPP unit at STUK. The role of the project group is to plan and co-ordinate the review work. The line organisation at STUK performs the actual review work to which the project group also participates. One specific task of the project group is to evaluate utility's quality management. After planning the review process, the duty of the project group is to see that the work performed at STUK proceeds as planned.

The FIN5 regulatory project at STUK is divided into 10 subprojects, which are introduced in Figure 2. One of the subprojects deals with radiation and environmental safety as well as emergency preparedness issues. It includes for example review of siting issues, radiation safety of the plant and related analyses, radiation instrumentation and emergency preparedness arrangements.

**Figure 2. Different sectors of the project group, which co-ordinate the license application process of the fifth Finnish NPP unit at STUK**



## Work planning and a tool for requirement management

From June 2002 to the end of 2003, the FIN5 project at STUK lived a so called preparation phase. The main task was the future work planning. A project manual was prepared, which includes for example a description of the project organisation, responsibilities and project management, main phases of the project and resource estimates. The planning was also performed on the subproject level. Every subproject manager made an inspection and review plan, which includes for example milestones for review process, resource allocations and prioritisation of items to review.

Another major task during the preparation was a development of a tool for Requirement Management. In Finland, the safety requirements of the nuclear power plants are introduced in the Nuclear Energy Act (990/1987) and Decree (161/1988), in five separate Decisions of Council of State (General Regulations for the Safety of NPPs, Physical Protection of NPPs, Emergency Response Arrangements at NPPs, the Safety of a Disposal Facility for Reactor Waste and the Safety of Disposal of Spent Nuclear Fuel). By virtue of the Nuclear Energy Act (990/87) and the Decision of the Council of State (395/91) on General Regulations for the Safety of Nuclear Power Plants, STUK issues detailed regulations concerning the safety of nuclear power plants. These regulations are called YVL Guides. In the preparatory work for Requirement Management system, the YVL Guide requirements that the licensee (applicant) and the new reactor have to fulfil were identified. Also the requirements for STUK's oversight were defined.

The first version of the Requirement Management tool was implemented with simple Excel files. The second step will be a more sophisticated database application, where the search of the data is easier. The requirement management system can be used for example as a standard review plan for a Preliminary Safety Analysis Report because all requirements are linked to the different Chapters of the SAR.

## Radiation safety related YVL guides

After the decision in principle (DIP), STUK made a plan according to which the existing YVL Guides were evaluated and updated. The guide YVL 7.18, concerning the radiation safety aspects in the design of NPPs, was up-dated during 2003. The main content of the new guide is shown in Figure 3. In this updated guide, accident situations including severe accidents and aspects of decommissioning of the plant are taken into account in more detail. Other relevant radiation safety guides during the construction license review phase are:

- YVL 1.10 Safety criteria for siting a NPP;
- YVL 7.1 Limitation of public exposure in the environment of and limitation of radioactive releases from NPPs;
- YVL 7.2 Assessment of radiation doses to the population in the environment of a NPP;
- YVL 7.3 Calculation of the dispersion of radioactive releases from a NPP;
- YVL 7.5 Meteorological measurements at NPPs;
- YVL 7.6 Monitoring of discharges of radioactive substances from NPPs;

- YVL 7.11 Radiation monitoring systems and equipment in NPPs.

Relevant safety guides during the operating license review phase are:

- YVL 7.4 NPP emergency preparedness;
- YVL 7.7 Radiation monitoring in the environment of NPPs;
- YVL 7.8 Environmental radiation safety reporting of NPPs;
- YVL 7.9 Radiation protection of NPP workers;
- YVL 7.10 Monitoring of occupational exposure at NPPs. [2].

**Figure 3. The main contents of the YVL guide 7.18 on the radiation safety aspects in the design of NPPs**

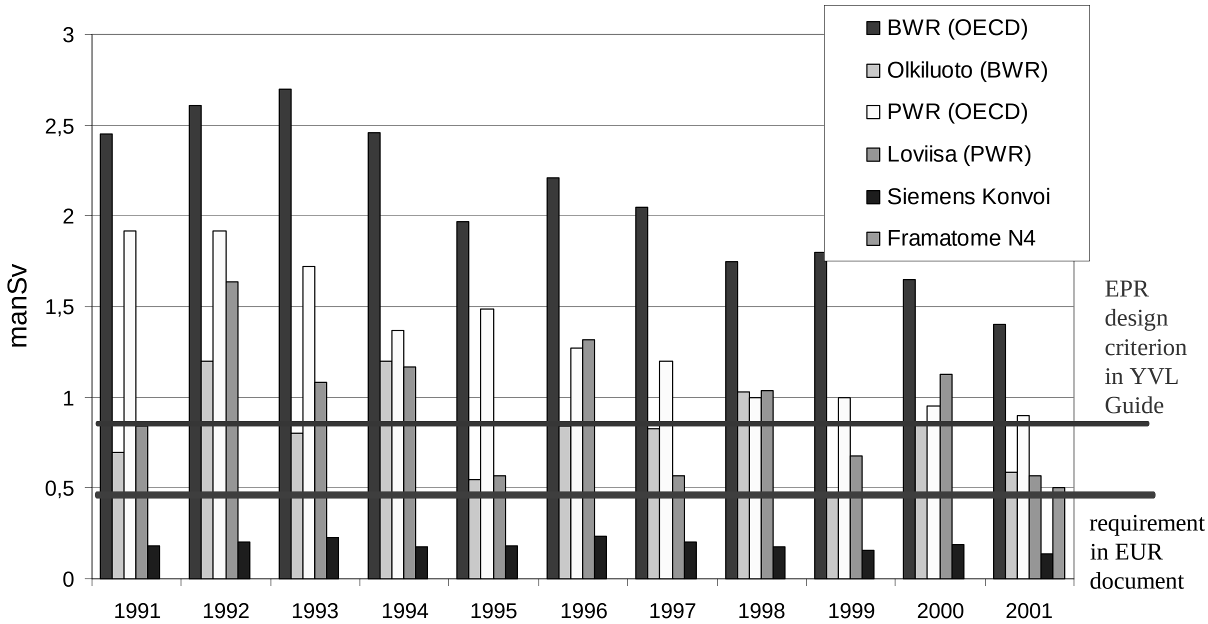
1 General
2 Design principles
    2.1 General requirements
    2.2 Radiation sources and shields
    2.3 Materials and their corrosion resistance
    2.4 Plant layout
    2.4.1 Rooms and access routes
    2.4.2 Entering and leaving the controlled area
    2.5 Decontamination of rooms and equipment
    2.6 Decommissioning
    2.7 Accidental situations
3 Radiation safety in systems design
    3.1 Individual systems and components
    3.2 Pipelines
    3.3 Drainage and leak collection systems
    3.4 Treatment of resins and concentrates
    3.5 Limitation of the effluent release
4 Regulatory control

### Collective dose target

In the updated regulatory guide YVL 7.18, a new design criterion for an annual personnel collective dose of 0.5 manSv per 1 GW of net electric power averaged over the plant life is set forth. Almost similar criterion is also written in the European Utility Requirements (EUR) document, where the target for annual collective effective dose averaged over the plant life is set as 0.5 manSv per reactor unit.

The existing reactors in Finland were commissioned between 1977 and 1981. Average personnel collective radiation doses per reactor for operating OECD country NPPs [3] and for existing Finnish NPPs for the years 1991-2001 are shown in Figure 4. The collective dose at the Olkiluoto NPP has been clearly under the international average value of the BWR reactors. On the other hand, the comparison of the collective dose at the Loviisa NPP to the average value of the PWR reactors does not give such an excellent result. Average collective doses per reactor of the German Konvoi generation NPPs (Emsland 1, Isar 2 and Neckarwestheim 2) and French N4 generation NPPs (Chooz B1 and B2, statistics only from the year 2001) [3] and the Finnish regulatory collective dose design criterion calculated for the EPR net electric power (0.8 manSv/year) and the collective dose target in the EUR document (0.5 manSv/year) are also shown in Figure 4. The statistics of the Konvoi NPPs would indicate that the collective dose in the EPR could be low.

**Figure 4. Average personnel collective radiation doses per reactor for operating OECD country NPPs, German Konvoi generation NPPs, French N4 generation NPPs and for existing Finnish NPPs**




## On-site habitability during accident situation

In a nuclear power plant, on-site habitability during accident situations has to be taken into account. “On-site habitability” determines conditions whether or not the occupancy of a certain area inside or outside the site buildings is possible on a continuous or transient basis. The regulatory guide YVL 7.18 requires analyses of the magnitude and location of the possible radiation sources and evaluation of doses in different accident management and emergency preparedness measures. In the design process, these doses shall not exceed the normal dose limits of a radiation worker. In a case of a real emergency situation, the normal dose limits can be exceeded while performing measures needed to save lives or restrict the radiation hazard and bring the radiation source under control.

Assessment of the on-site habitability during severe accidents at the existing Finnish nuclear power plants has been primarily done during 1980s and 1990s. A reassessment was done in 2002-2003 [4]. The method for assessing habitability included the following steps: defining the accident scenario and the sources of radiation, identification of the possible severe accident management actions and vital areas of the plant and finally calculating the dose rate levels in these vital areas. Habitability was evaluated based on the calculated dose rate levels, the occupancy times and the dose limits. Radiation hazard was classified into three parts, i.e., possible direct radiation from the containment, air contamination and systems carrying radioactive air or water. The results showed that direct radiation from the containment is generally adequately shielded but penetrations and hatches have to be separately analysed and the radiation dose levels near them are usually rather high. Skyshine radiation from the reactor containment is a special feature at the Loviisa NPP and the nearby area outside the buildings might have very limited access for the first hours after the accident. The skyshine effect is not usually relevant hazard in nuclear power plants, because they have adequate concrete shielding also in the roof of the containment. An interesting result was that air contamination also in the building next to the containment might be a hazard even if the containment is intact and leaks only at the nominal rate. Systems outside the containment can also create higher local radiation levels, e.g. near the emergency core cooling systems, containment spray system, sampling systems and containment filtered venting system.

**References**


1. Ministry of Trade and Industry (2002), *Nuclear Energy in Finland*, Energy Department, Helsinki, October.
2. http://www.stuk.fi/english/publications/yvl-guides.html
3. CEPN (2001/2002), MADRAS database, version 3.8 (Rev. 9), European Technical Centre.
4. Alm-Lytz, K. (2003), *On-Site Habitability at Finnish Nuclear Power Plants during Severe Accidents*, Licentiate's thesis, Helsinki University of Technology, Espoo.

## THE FIFTH NUCLEAR POWER PLANT IN FINLAND FROM THE RADIATION PROTECTION POINT OF VIEW

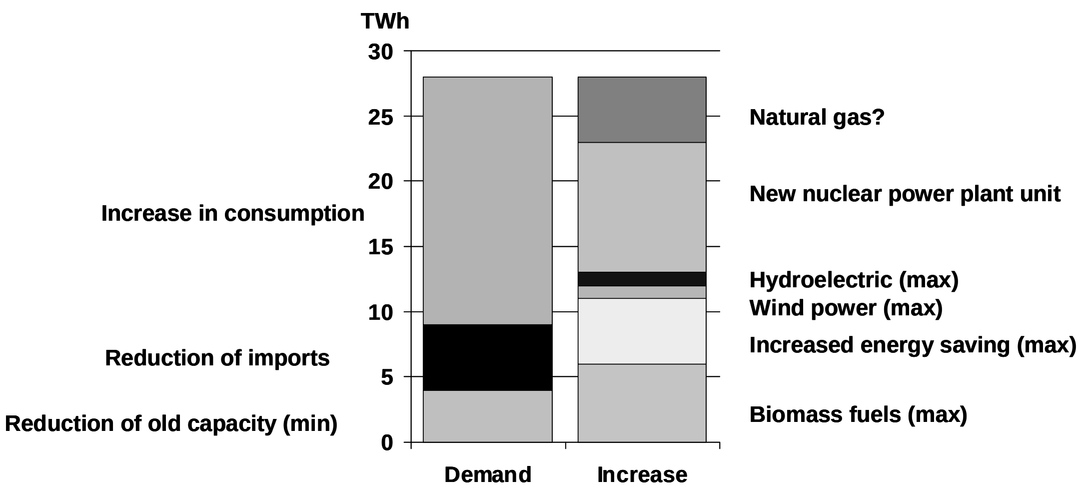
**J. Sovijärvi**
Teollisuuden Voima Oy, Finland

### Introduction

Teollisuuden Voima Oy (TVO) is a private own company, which operates two ASEA Atom designed BWR units in Olkiluoto island in the west coast of Finland. TVO is founded in 1969 and the commercial operation of units OL1 and OL2 started in 1979 and 1982 respectively. After two power increase projects the current net power of 840 $MW_e$ per unit (corresponds to 2 500 MW thermal power) was achieved in 1998. The company produces electricity to its shareholders at cost.

During the year 2003 over 25% of the electricity in Finland was produced by nuclear power, meanwhile the share of import was 5.7%. Other notable manners of electricity production are hydropower, co-generation (district heating and industry) and condensing power (fossil fuel). Up to the year 2015 the annual electricity demands are estimated to increase with 25-30 TWh. This is due to the consumption increase, estimated decrease of import and reduction of old capacity (Figure 1).

**Figure 1. Additional electricity demand and production increase until 2015 (This estimation was originally released in 1999)**

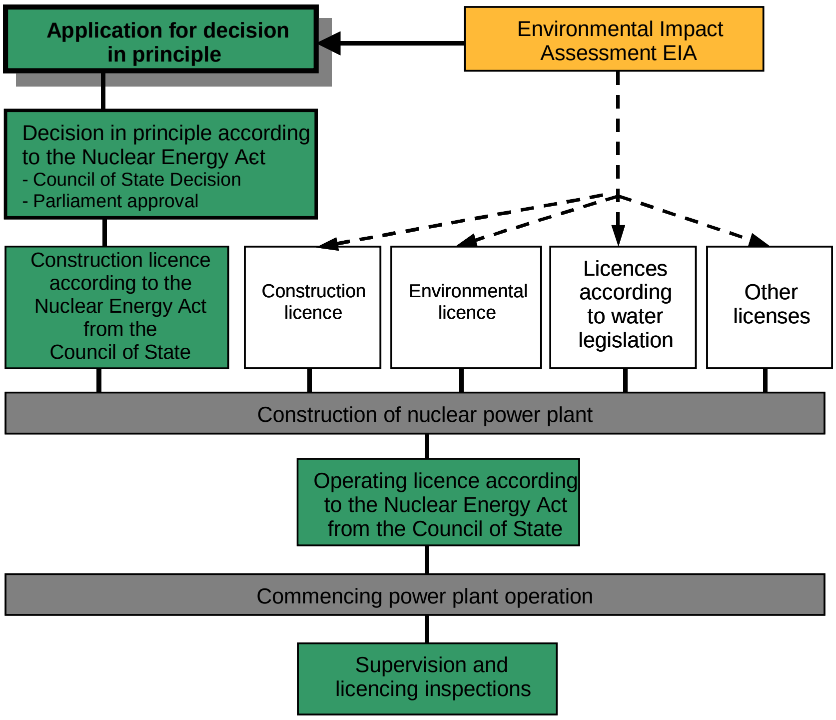



Source: Council of State National Climate Strategy and Finergy

**Overall view on the nuclear power plant licensing and procurement procedures**

The licensing schedule is shown in the Figure 2. In the late 90s the preparedness phase was carried out. The Environmental Impact Assessments (EIA) for two alternative sites, Olkiluoto and Loviisa were issued. During this phase a number feasibility studies for potential light water reactor concepts were also done.

**Figure 2. Summary of the licensing procedure of the new nuclear power plant**



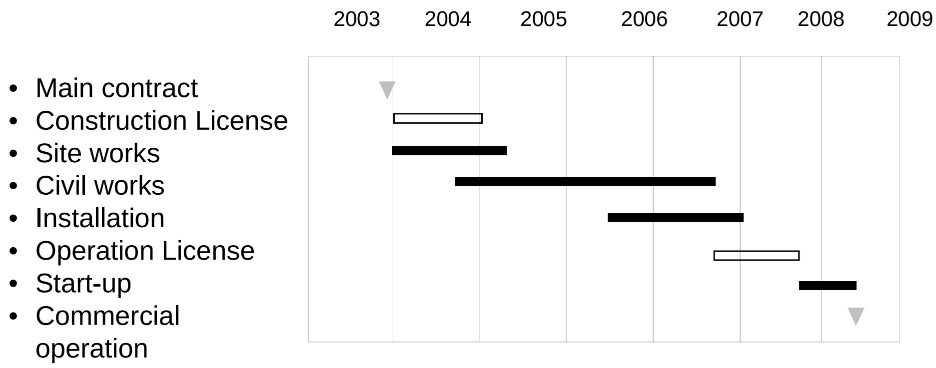

TVO put in the application for decision in principle (DIP) in the autumn 2000. The Council of State (i.e. Finnish government) accepted the application 17 January 2002 and the Finnish Parliament confirmed the decision after voting 24 May 2002 (107 in favour, 92 against). The decision in principle noted that the construction of a new light water reactor, PWR or BWR, on either of the two existing nuclear power plant sites in Finland, Loviisa or Olkiluoto is in accord with the "overall good of the society". The electric output of the new unit is limited to 1 000-1 600 $MW_e$.

Bid invitations were sent out on 30 September 2002 and the bids were received on 31 March 2003. During the bidding phase a number of bids by different power plant suppliers were evaluated.

Olkiluoto was selected as the location of the new power plant in October 2003. The turnkey contract with consortium Framatome – Siemens was established in December 2003.

The execution phase started in the beginning of the year 2004. The commercial operation of the plant is planned to start before the end of this decade. The overall time schedule of the process is shown in the Figure 3.

**Figure 3. Olkiluoto 3 overall schedule**




## Bid evaluation phase: radiation protection aspects

About 100 persons participated in the evaluation work in total, half of them full-time and the rest part-time. Special attention was paid on the correct handling of confidential and secret information. E.g. evaluation group has own data network, which was physically separated from TVO's administrative network and from the outside world. Eight working groups were formed to carry out the technical evaluation of the received bids:

- contract;
- fuel contract;
- technical group;
- scope of delivery;
- reactor core;
- safety;
- calculation;
- operation and maintenance.

Most of the radiation protection issues were handled in the operation and maintenance group, which also was in charge of evaluation of O&M issues, waste handling, chemistry and outage performance as well. Certain questions were also co-operated with other working groups, especially Safety group and Technical group.

The call for bids, with respect to radiation protection, was based on the EUR-document (European Utility Requirement). EUR-document was however reviewed and completed with further demands from YVL-guidelines, other relevant national requirements, TVO's own expertise and operation experience.

The most essential requirement is the minimisation of both collective and individual doses. As stated in the ICRP publication 60, BSS 96/29/Euratom, YVL 7.9 etc. the individual radiation exposure of workers is limited to 50 mSv/year and to 100 mSv/5 years. Taken into account the decreasing international trends and the possibly strengthened recommendations and limits in the future the maximum individual dose target in Olkiluoto 3 was set to 5 mSv/year. The collective annual dose target given in the YVL 7.9 is less than 2.5 manSv/$GW_e$ as two years running mean for operating units. YVL 7.18 states that the target for planning of new NPP units is less than 0.5 manSv/$GW_e$/year (life time mean). TVO defined the target value of the annual collective dose to be less than 0.5 manSv for Olkiluoto 3 (life time mean). During accidents the maximum allowable dose for workers, who are not performing life saving actions is set to 50 mSv and the maximum allowable dose for public during normal operation and anticipated incidents is less than 0.1 mSv (site specific value).

Some key issues, which have a significant contribution on the effectiveness of radiation protection are shieldings, plant lay-out, placing of RP-facilities, source term minimisation, corrosion, deposition of corrosion and fission products, water chemistry, fuel handling, maintenance activities in the controlled area and in-service inspections.

As a result to the bid evaluation the power plant unit equipped with EPR (European Pressurised Water Reactor) nuclear island and Siemens’ turbine island was chosen. The radiation protection aspects, which were taken into account during the evaluation process were not deciding. All plants bid could be acceptable at least after some modifications. The EPR concept is based on the newest German and French PWRs. Framatome ANP names both N4-generation (Civaux 1 and 2, Chooz B1 and B2) in France and the Konvoi-generation (Neckarwestheim 2, Emsland, Isar 2) in Germany as references.

**Main activities during the execution phase and conduct to operation**

One of the most important tasks before the start up is the developing enough competence and understanding on the behaviour of the new unit. TVO has a lot of experiences on operation of BWRs and has during its history managed to keep both individual and collective exposures relatively low. An extremely important and challenging objective is to receive equal results in the future in OL3 as well.

Most of the employees have worked since 70s or 80s in TVO and many of them will retire within the next ten years. Obviously the change of generation makes further challenges to everybody in the company. Up to the present the interest in working in the nuclear power company has been sufficiently high. Radiation protection organisation, which is today common for all nuclear facilities in Olkiluoto will certainly expand.

During this year the focus is in the licensing process and generating a good co-ordination between all parties attending the project. For example the collection of YVL guides are continuously under updating but none NPP have been licensed in accordance with them before.

Later the most resources will be needed for commissioning and procurement of systems and equipments not included to the turnkey delivery.

# MODERNISATION OF THE ACCIDENT LOCALISATION SYSTEM AND RELEVANT DOSE EXPOSURE ON UNIT 4 OF KNPP

**G. Valtchev, M. Neshkova, A. Nikolov**
Nuclear Power Plant Kozloduy, Bulgaria

## Abstract

In 2001 a modernisation of the accident localisation system (ALS) on Unit 4 was accomplished. The outage duration was longer then usually and special dose budget was elaborated. All ALS work was performed by external organisation. An ALARA implementation was recognised priority. The really accumulated collective doses were analysed and conclusions drawn. A short film on CD was prepared.

## Preliminary information

According to the modernisation programme' 97 of NPP Kozloduy priority task was the accident localisation system (ALS). Main part of it is the Vortex Jet Condenser.

Due to modernisation activities in 2001 including the modernisation of accident localisation system the outage of unit 4 was prolonged to 4 months, (22.08.2001-22.12.2001). The projected collective dose was 1 610 man.mSv. The projected collective dose (dose budget) just for ALS activities was 300 man.mSv. This dose budget was based on the technology to be used and the expected working time for the main activities. Consideration was given to the gamma dose rate (dose rate mapping) as well.

The gamma dose rate on the working places in the restricted control area (RCA) was 0.005-0.03 mSv/h. For a short-time activities, on some hot spots the dose rate was up to 0.80mSv/h.

Before the work started a group of engineers and worker was delegated to Novovoronezh NPP, where similar system was already erected. The purpose of the visit was to gather information about:

- what instrumentation was used;
- what facilities should be available and applied;
- what was the most crucial task;
- how the work was scheduled;
- what is the optimum number of workers per shift;
- radiation safety measures;
- any experience feedback.

**ALARA implementation**

The project was performed by subcontractor. The modernisation activities started from the first days along to the end of the outage.

The existing ALARA approach was implemented:

- instruction of the personnel about used technology and systems location;
- gamma dose rate mapping at the entrance of the RCA;
- pre-briefing of workers;
- all preparatory work was performed outside the RCA;
- decontamination of the floor;
- individual dose control by TLD and alarm electronic dosimeters;
- weekly management meetings.

A short ALARA course was given to each worker group. A very detailed planning of the work sequence of the different tasks and transportation routes were established.

One key aspect was the effective empowerment of the workforce. The work management of this modernisation emphasised this aspect and we believe it was successfully applied. On daily and weekly meetings we tried to talk to the workers and third line management, and motivate them to perform job in efficient and effective manner. Good communication with RP personnel and all workers, and everyday dose records reports helped to follow the set goals.

Monitoring of the internal contamination was performed before and at the end of the activities.

**Dose management**

Dose management was conducted according to the procedures in KNPP and ALARA meetings decisions:

- dosimetry with TLD badges and electronic alarm dosemeters were used for the dose control of the personnel;
- the highest permissible dose in the Radiation Work Permit was 0.5 mSv;
- the maximum permissible accumulated individual dose for all ALS modernisation 5 mSv;
- a daily analysis and comparison of the expected and committed dose;
- dose rate mapping was performed in any single RWP;
- continuous monitoring of the levels of air contamination.

Dose reports were accomplished on daily basis and transmitted to all involved persons.

## Results

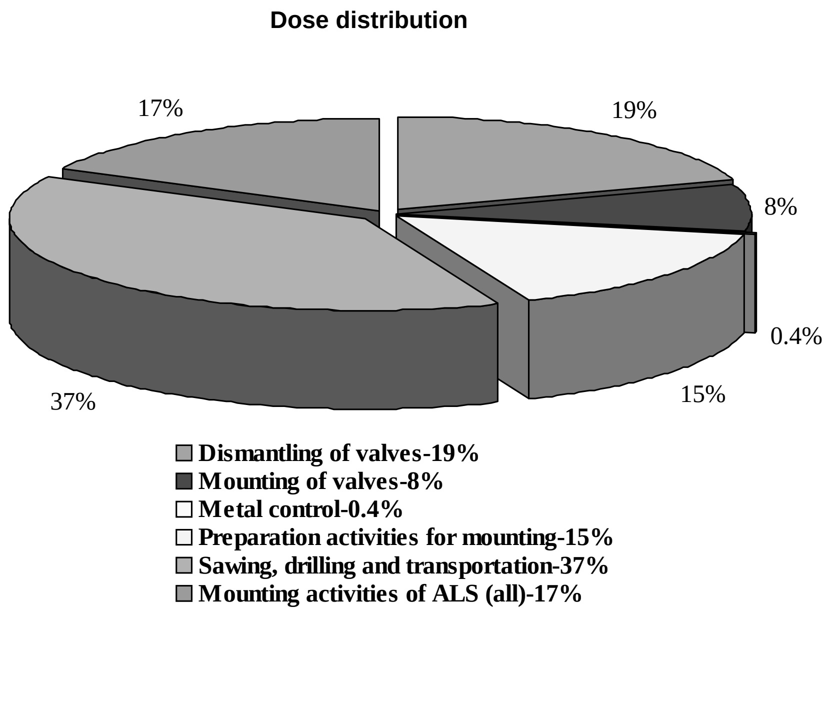
### *Dose analysis*

In the end of ALS modernisation about 606 radiation work permits (RWP) were issued. About 210 workers were controlled. No one worker received a dose higher then 0.2mSv/day. The maximum individual dose was 5.8mSv, about 15% higher than the allowable value. The worker was busy with cutting, and dismounting activities. The average individual dose due to ALS modernisation was 1.16mSv. The real collective dose for all ALS activities was 243.4 man.mSv, and for the whole outage – 1 732.7 man.mSv. About 35 different tasks were observed.

No internal contamination at the end of the ALS modernisation was detected. Based on the requested task in the radiation work permits, dose distribution analyses were performed. All RWPs were grouped in 6 main tasks:

- dismantling of valves – 123 RWP with collective dose 47 man.mSv;
- mounting of valves – 31 RWP with collective dose 20.1 man.mSv;
- metal control – 11 RWP with collective dose < 1man.mSv;
- preparation activities for mounting – 67 RWP with collective dose 37.8 man.mSv;
- sawing, drilling and transportation – 254 RWP with collective dose 90.2 man.mSv;
- mounting activities of ALS (all) – 96 RWP with collective dose 42 man.mSv;

The dose distribution is shown on the following chart.



### *Radioactive waste generation*

During the reconstruction activities a quite big amount of concrete and metal wracks was cut out and removed from the restricted area. Most of it, about 200 $m^3$, was not contaminated. The radiological limitation was: non fixed beta activity less then 0.4 $Bq/cm^2$, and gamma-dose rate less than 0.20 μSv/h.

All measurements were performed on previously selected area with low background (15-20 μSv/h). About 5 $m^3$ were determined as radioactive waste.

## Conclusion

Good work management and a first attempt of effective empowerment of the workers gave satisfactory results.

Although the work was not typical, and performed for a first time, the ALARA implementation reduced the projected collective dose with 19%.

## References


1. IAEA (2002), *Optimization of Radiation Protection in the Control of Occupational Exposure*, Safety Report Series No. 21, Vienna.
2. IAEA (2001), *Second EC/ISOE Workshop on Occupational Exposure Management at NPPs*, Working Material, Vienna.

# SESSION II

# HIGHLIGHTS OF EPRI RADIATION EXPOSURE MANAGEMENT PROGRAMME

**S. Bushart, D. Hussey**
EPRI, United States of America

## Abstract

Radiation exposures at US nuclear power plants continue to decline, but radiation protection engineers face increasing challenges as a result of shorter outages, core up rating and remedial measures to mitigate materials degradation. This paper describes recent technology advances that have been implemented to control out-of-core radiation dose rates.

The use of noble metal chemical application to mitigate core internals cracking in BWRs resulted in increase in radiation fields at some plants as a result of redistribution of activated corrosion products. This paper describes the investigation of corrosion transport processes that led to successful recommendations to control fields.

Zinc injection has been implemented at several PWRs, resulting in ~20% reduction in radiation fields per cycle. This paper outlines work to optimise zinc injection to maximise the dual benefits of reduced stress corrosion cracking and radiation control, while avoiding adverse side effects.

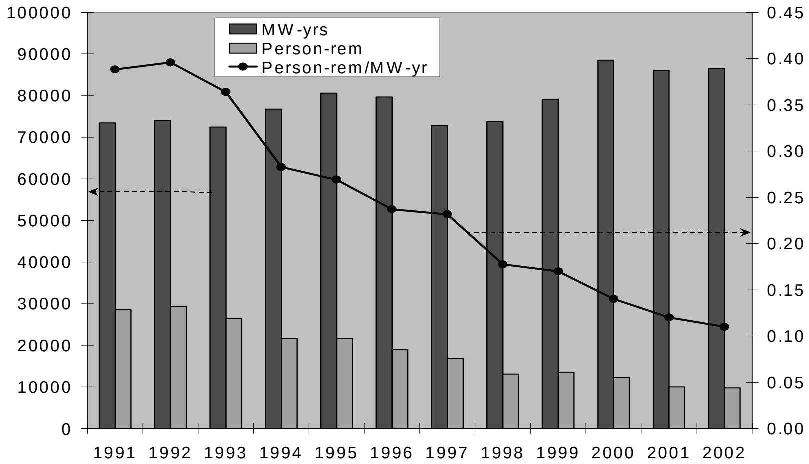
A patented technique for the removal of activated corrosion products from fuel cladding using ultrasonic cleaning has been implemented at several PWRs, and is currently being qualified for BWR applications. The latest plant data concerning this technology are presented.

The EPRI Radiation Field Control Manual provides RP managers, engineers, and chemists with a valuable reference to the current field control and reduction technologies that are employed in the United States nuclear power plants. This paper discusses briefly the material described within the document.

## Introduction

Occupational radiation exposures at light water reactors worldwide have decreased in a trend extending over two decades. Median collective exposures at US plants and electric generation data, shown in Figure 1, indicate that the collective exposure per MW electricity generated has dropped by an order of magnitude in the last 20 years. Despite this outstanding success story, challenges remain, as plants age, output is increased and outages become shorter. Accordingly, EPRI has actively pursued advanced technology to reduce out-of-core radiation fields, with recent developments described in this paper.

**Figure 1. Collective radiation exposure and electricity generated at US nuclear power plants**




## BWR noble metal chemical application

BWRs implement hydrogen water chemistry to reduce electrochemical potential and mitigate cracking of core internals. In the 1990s, hydrogen concentrations were increased to provide more protection in the reactor vessel. This chemistry change resulted in increased out-of-core radiation fields; most plants introduced depleted zinc injection to control dose rates that increased from the increased hydrogen rates.

Main steam line radiation from $^{16}N$ activity increases under HWC conditions because the nitrogen species formed under reducing conditions are more volatile than in oxidising environments. The magnitude of the effect increases at higher hydrogen concentrations. Studies at General Electric showed that the presence of noble metals on structural materials would significantly reduce the hydrogen concentration required to achieve the IGSCC protection potential of -230mV(SHE). Noble metal chemical addition (NMCA) was introduced at Duane Arnold BWR as an *in-situ* method of reducing the amount of hydrogen required to lower the ECP on material surfaces, which would also mitigate the effects on operating radiation fields.

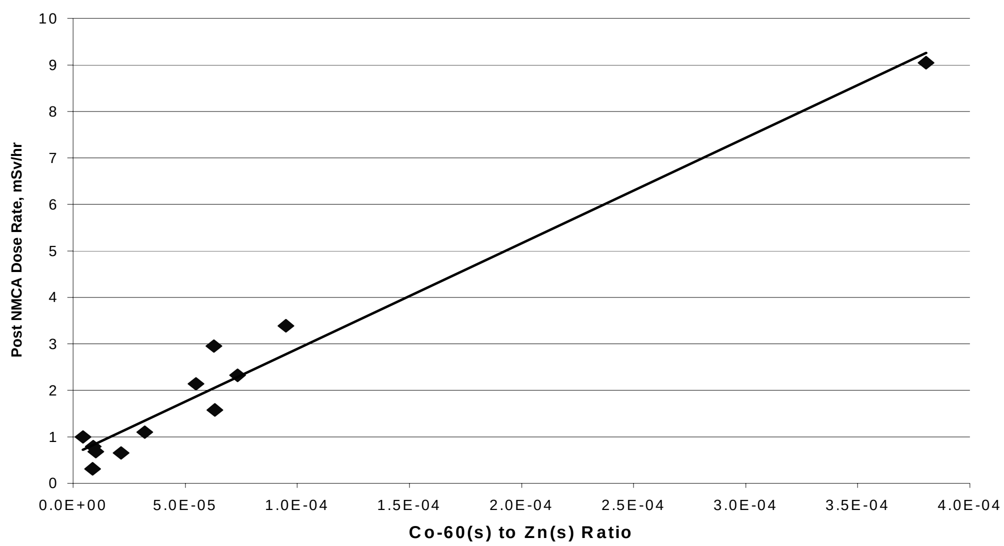
One method of achieving the ECP specification, Noble Metal Chemical Application (NMCA), was developed to avoid increased dose rates and high hydrogen usage. NMCA deposits very small amounts of platinum and rhodium metal on the wetted surfaces within the reactor vessel and reactor coolant system. These noble metal deposits catalyse recombination reactions of hydrogen with $O_2$ and $H_2O_2$ at these surfaces. Protective ECPs are achieved when the molar ratio of hydrogen to total oxidant in reactor water reaches a value equal to or greater than two. In the BWR, the molar ratio reaches the value of two at very low feedwater hydrogen addition concentrations (usually between 0.1 and 0.15 ppm). Also, there is little or no increase in main steam line radiation from $^{16}N$ activity at these hydrogen addition levels.

During the initial implementation of Moderate HWC to the BWR fleet, it was noted that the introduction of feedwater hydrogen significantly increased shutdown dose rates at some plants at the end of that fuel cycle. However, the shutdown dose rate effects of hydrogen water chemistry can be mitigated by feedwater zinc addition. Similar effects are found with NMCA.

In the first cycle after NMCA application, all plants have seen an increase in both soluble $^{60}Co$ (1.4X to 3X increase) and insoluble $^{60}Co$ (2X to 50X), resulting mainly from the release of material from fuel surfaces. Since incorporation of $^{60}Co$ on reactor surfaces usually contributes 80 to 90% of the shutdown dose rate, it may be surprising that dose rates at all NMCA plants have not increased significantly. However, testing during the qualification phase of NMCA in the mid 90s showed that noble metal treated coupons experienced greatly reduced pickup of $^{60}Co$ when exposed under simulated BWR conditions with both hydrogen and zinc additions in the water. Thus, even with the increase in reactor water $^{60}Co$ under post NMCA conditions, the addition of zinc at 5 ppb or greater can offset the tendency for increased dose rates. Those plants that had well established zinc injection programmes and increased feedwater zinc to maintain 5 ppb or greater in the reactor water immediately after NMCA had unchanged or lower post NMCA dose rates.

The observed increases in reactor water $^{60}Co$ and other isotopes are similar to those that occur when a plant first initiates standard HWC, except that the effects are magnified with NMCA. When the environment is changed from oxidising to reducing, there is a change in the stable form of oxide, resulting in a conversion of hematite ($Fe_2O_3$) to magnetite ($Fe_3O_4$). This change releases both soluble and insoluble species from fuel surfaces. With NMCA, this change occurs over all treated surfaces at the same time, creating a long-lasting increase in soluble and insoluble $^{60}Co$ and other species. As with the shutdown dose rate increase seen with HWC, zinc in the reactor water mitigates the dose rate increase following NMCA, both by suppressing the release of $^{60}Co$ from fuel deposits and by competing with cobalt for the same tetrahedral crystal sites in spinel corrosion films.

**Figure 2. Measured post NMCA shut down dose rates vs. the reactor water $^{60}Co(s)$ to Zn(s) ratio**



Theory predicts that because zinc ions and $^{60}Co$ ions compete for the same space in the lattice of spinel corrosion films and crud, the lower the ratio of reactor water $^{60}Co$ to reactor water zinc, the less $^{60}Co$ will be incorporated into new films forming during the restructuring process that occurs in post NMCA operation. The trend line in Figure 2, which plots the post NMCA shutdown dose rate versus the reactor water $^{60}Co$ to Zn ratio.

Clearly, the lower the ratio, the lower the subsequent shutdown dose rate, as predicted by theory. Even though in all cases the soluble $^{60}Co$ level increased in post NMCA operation, those plants that maintained a zinc reactor concentration in the 5 to 10 ppb range saw either no change or a decrease in dose rates. In summary, the post NMCA dose rates are primarily controlled by the ratio of reactor water $^{60}Co(s)$ to Zn(s). Other secondary factors, such as feedwater iron level and length of time prior to the NMCA application that HWC and feedwater zinc injection have been employed also play a role.

**PWR zinc injection**

The current trend toward longer fuel cycles in PWRs has placed an added concern on optimisation of RCS chemistry. Despite application of an optimum chemistry control programme, higher radiation fields may be observed. The reduced frequency of outage-related work, along with proper work planning and application of shutdown chemistry controls may decrease the impact of longer fuel cycles.

Laboratory studies, increasingly complemented by experience in operating PWRs, indicate a benefit of zinc additions to the reactor coolant system as a means to effect dose rate reductions and potentially mitigate the occurrence and severity of primary water stress corrosion cracking of Alloy 600.

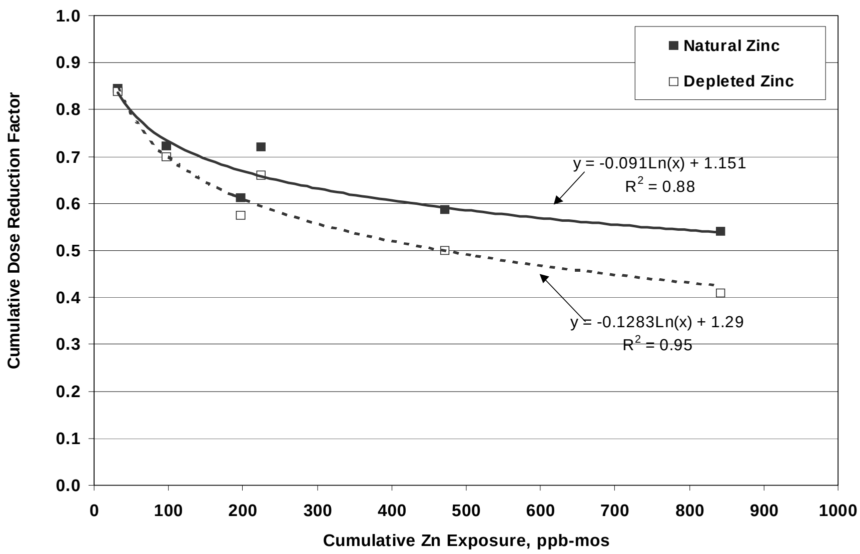
EPRI and Southern Nuclear cosponsored the initial field demonstration of zinc addition at Farley Unit 2 in 1994-95. The results of this demonstration confirmed the beneficial effects of zinc in mitigating radiation fields, which is now well established with positive results observed in both domestic and German PWRs. A more elusive issue has been the effectiveness of zinc in mitigating PWSCC. Although laboratory testing has indicated a beneficial zinc effect on mitigating crack initiation in Alloy 600, data for a beneficial effect on crack propagation are mixed. Accordingly, EPRI has initiated a comprehensive laboratory study to quantify the effects of zinc on primary water stress corrosion cracking (PWSCC).

Measurements of dose rates after Cycle 10 at Farley-2 with zinc addition showed a reduction of 24% at steam generator channel heads of which 11% was attributed to zinc addition and shutdown chemistry practice. Zinc-65 was less than 10% of the radioisotopic mix and only a minor contributor to the radiation fields. A dose saving of 40 man rem per fuel cycle was estimated for Farley-2 after five cycles of natural zinc addition. No zinc was added during Cycle 11 at Farley-2. Farley-2 resumed zinc addition in Cycle 12. However the period of zinc addition during Cycle 12 was too short (3 months) to assess its effect on either dose rates, fuel cladding corrosion, or PWSCC.

Diablo Canyon Unit 1 began adding natural zinc in June 1998, followed by Unit 2 in 1999. After the first application on Cycle 9, the levels of $^{58}Co$ in the reactor coolant increased significantly in both Unit 1 and Unit 2. Only modest activity increases were observed following Cycle 10, the second with zinc injection. The data suggest that at least one cycle with zinc chemistry is required to stabilise the zinc-substituted corrosion product deposits. Zinc injection reduced shutdown dose rates for both Diablo Canyon units. After the second cycle with zinc chemistry, steam generator dose rates were approximately 42% lower than levels prior to zinc injection at Unit 1 and 59% lower at Unit 2, although other factors may have contributed to this improvement. The activity ratios of coolant particulates for the second cycle suggest that the corrosion product deposits were stabilised through the incorporation of zinc.

Several PWRs in United States and Europe are currently injecting zinc into the primary coolant, at levels ranging from 5 to 30 ppb, with the higher levels selected to mitigate PWSCC. The data from these plants indicates that zinc addition continues to lower shutdown dose rates, but the reduction is less for each succeeding fuel cycle. This trend is reasonable, since the corrosion films are becoming conditioned with respect to exchange of nickel and cobalt for zinc. The trend seems to correlate with the cumulative exposure to zinc, as shown in Figure 3, which was developed by Westinghouse for EPRI in a published EPRI report. Plants that used depleted zinc, to avoid activation to zinc-65, showed the greatest improvement.

**Figure 3. Cumulative dose reduction factor as a function of cumulative zinc exposure**




Zinc addition to the primary coolant, even at the relatively low levels of approximately 5 ppb in the RCS, appears effective for reducing radiation dose rates, and zinc addition is being used at several plants for this purpose. Zinc is used at the 20-50 ppb level at Farley and Diablo Canyon to mitigate PWSCC. Consideration should be given to performance of a fuels evaluation before addition of zinc at these higher levels, particularly for higher rated cores.

The 2003 edition of the EPRI PWR Water Chemistry Guidelines recommends that each PWR consider injecting zinc. In preparing both equipment and documentation for injection at low levels for radiation control, it is prudent to plan ahead for higher injection rates if the ongoing laboratory programme confirms the benefits of zinc in mitigating PWSCC.

**Ultrasonic cleaning of nuclear fuel**

When PWRs operate with higher fuel duty and longer cycles, sub-cooled nucleate boiling in the upper fuel spans is a consequence. Thermodynamic and hydraulic factors favour deposition of corrosion products on the boiling surfaces of the fuel, resulting in axially non-uniform deposition on high-duty fuel. Axially variable distribution of boron compounds in these fuel deposits is an important cause of local flux depression, termed axial offset anomaly (AOA).

Ultrasonic fuel cleaning was demonstrated to be an effective means for removing PWR fuel deposits, hence mitigating the AOA problem. In addition the reduced fuel crud inventory was shown to reduce dose rates on subsequent shutdown for refuelling.

Although ultrasonic fuel cleaning has been applied at PWRs primarily for mitigation of AOA, a reduction in ex-core dose rates, and consequently personnel exposure, is also observed. These ALARA (as low as reasonably achievable) benefits can be advantageously achieved at BWRs as well. The Callaway PWR has observed dose rate reductions on the order of 50% for an outage following operation with cleaned reload fuel. Such reduced radiation fields can have a significant favourable impact on personnel dose. EPRI-sponsored modelling calculations by Westinghouse for PWRs and General Electric for BWRs have confirmed that such dose reductions should be expected if the corrosion products can be effectively removed from reload fuel.

Following an extensive qualification programme, fuel reliability following ultrasonic cleaning was first demonstrated at the Callaway PWR, where no fuel failures attributable to the ultrasonic process have occurred in the sixteen lead test assemblies, nor in two fuel cycles of fully cleaned reload fuel. At the time of writing, the technology has already been used subsequently at three other PWR plants in United States, and the first BWR application is planned for 2004.

Callaway has experienced AOA for many of its recent fuel cycles, and the plant staff has worked actively to mitigate this problem. The data for Cycle 12, for which all reload fuel has been cleaned, indicate that fuel cleaning is of significant value in controlling AOA. A reduction in ex-core dose rates was a welcome secondary benefit. Based on the results described above, AmerenUE cleaned all reload fuel again prior to loading the core for Cycle 13 in Autumn 2002. Data on subsequent radiation fields show a significant reduction from pre-fuel cleaning (Cycle 10) to the most recent Cycle 12 (Figure 4). Although BWRs do not suffer from AOA, it is anticipated that ultrasonic fuel cleaning to remove crud from the fuel cladding surfaces will have several advantages:

1. Mitigation of potential crud-related fuel problems, especially for plants with high iron levels.
2. Removal of the largest source of $^{60}Co$, reducing reactor water concentrations of this isotope and resulting in lower radiation fields and reduced demand for depleted zinc.
3. Fuel cleaning after NMCA will remove high concentrations of noble metal from the fuel surfaces, allowing higher loading of noble metals on core internal surfaces.

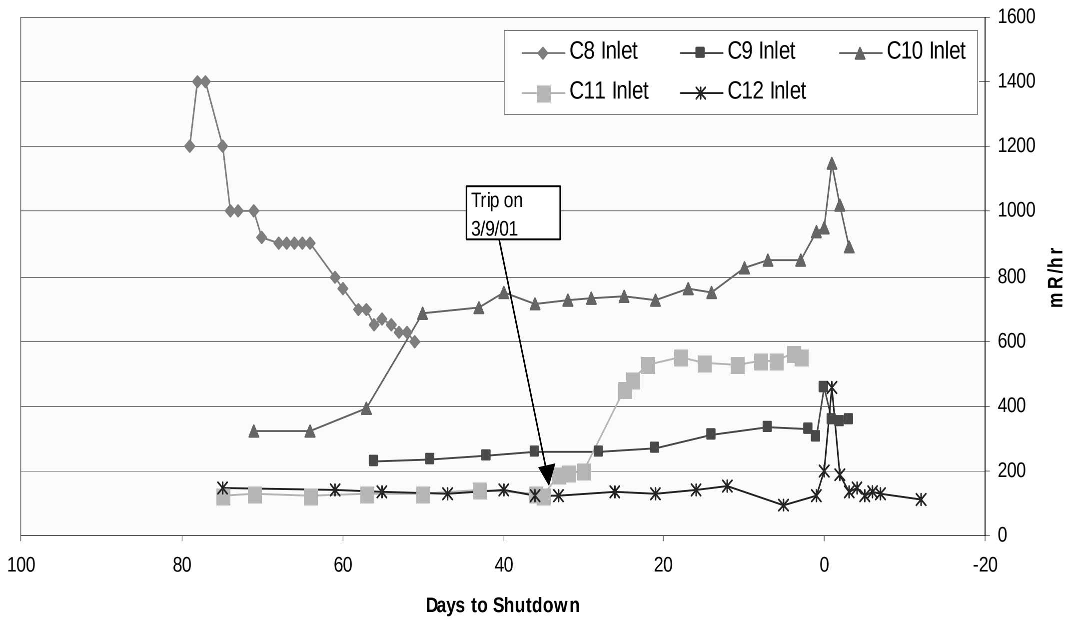
The above points suggest that ultrasonic fuel cleaning can be of real benefit for BWRs, especially if cleaning occurs immediately following NMCA application. The relatively high concentration of noble metals residing on the fuel after conventional NMCA application appears to serve as a source term for replacing the non-fuel noble metals as the latter are eroded or eluted from the plant components while in service. If such replacement redistribution from fuel to non-fuel surfaces has been a significant mechanism, the presence of noble metals on fuel may significantly increase the minimum interval between reapplications of NMCA. It may therefore be necessary to use higher noble metal concentrations initially to avoid more frequent reapplication of NMCA if the fuel is cleaned immediately following NMCA.

The option of cleaning BWR fuel was first considered (but not used) at River Bend in 1999. Mockup and laboratory tests were conducted, forming the basis of a preliminary conclusion that fuel cleaning could be an effective method for recovering highly-crudded BWR fuel assemblies.

As part of the final design qualification process, two sections of a fuel rod discharged after three cycles with NMCA were ultrasonically cleaned in the Vallecitos Nuclear Center hotcells during June 2003. Overall, no negative impact of ultrasonic energy on fuel pellets was observed.

A BWR fuel cleaner was designed in 2003; with the goal of constructing a prototype cleaner for demonstration tests in the Quad Cities fuel pool early 2004, followed by cleaning of reload fuel during the Quad Cities 2 refuelling outage in February 2004.

**Figure 4. Callaway letdown heat exchanger inlet dose rates**




## Radiation field control manual

The EPRI Radiation Field Control Manual has been used as a reference by RP managers, chemists, engineers and executives for many years. The manual was last updated in 1997, and several new technologies and methodologies have been developed and employed since then. A new version of the manual is to be written with few references to the previous manuals. The topics discussed in the manual will include:

- Radiation field origins and countermeasures – A review of radiation field sources, and an overview of the methods to prevent and reduce them.
- Source term reduction – Discussing the development and application of hardfacing alloys (such as EPRI's NOREM) in valves and control rod blades, as well as reviewing the impact of low-cobalt steam generator tubing on fields.
- Surface preconditioning – A review of the development and effects of surface pretreatments such as electropolishing, pre-oxidation, and EPRI's Stabilised Chromium Process (SCrP). New information will include the effects of electropolished steam generator channel heads.
- Effects of PWR primary chemistry on radiation fields – A review of the various primary coolant chemistry strategies is discussed, as well as the effects of zinc addition on fields is discussed. A discussion of enriched boric acid will also be included.
- BWR coolant chemistry effects on radiation fields – A review of the interactions and results of hydrogen water chemistry, natural and depleted zinc oxide addition, and noble metals chemical application is discussed.
- Chemical decontamination – A variety of dilute chemical decontamination (DCD) techniques and their various applications will be described.
- Ultrasonic fuel cleaning – The impacts of ultrasonic fuel cleaning on ex-core surfaces, as well as an introduction to the exposure risks involved in the waste disposal, is presented.
- Appendices describing the Quad Cities and Brown's Ferry Unit 1 restart dose reduction initiatives will be added as practical demonstrations of the concepts described in the manual.

The Radiation Field Control Manual will be published and available to EPRI members in December, 2005.

**Conclusions**

The three technologies described in this paper have been successfully introduced at operating plants. These examples demonstrate the close interaction between mitigation of materials degradation, fuel performance issues and radiation exposure concerns. It is interesting to note that these advances take advantage of synergistic benefits, producing win-win situations:

- The combination of HWC/NMCA/Zinc in BWRs mitigates stress corrosion cracking and reduces radiation fields.
- Zinc in PWRs reduces radiation fields and appears to mitigate PWSCC.
- Ultrasonic fuel cleaning addresses fuel performance issues in both PWRs and BWRs, and also reduces radiation fields.

In fact, these combinations of benefits facilitate the introduction of new technology, which is sometimes difficult to justify economically on radiation exposure grounds alone.

## COMPARISON OF PERFORMANCE INDICATORS OF DIFFERENT TYPES OF REACTORS BASED ON ISOE DATABASE

**H. Janžekovič, M. Križman**
Slovenian Nuclear Safety Administration, Slovenia

### Abstract

The optimisation of the operation of a nuclear power plant (NPP) is a challenging issue due to the fact that besides general management issues, a risk associated to nuclear facilities should be included. In order to optimise the radiation protection programmes in around 440 reactors in operation [1] with more than 500 000 monitored workers each year [2], the international exchange of performance indicators (PI) related to radiation protection issues seems to be essential. Those indicators are a function of a type of a reactor as well as the age and the quality of the management of the reactor. In general three main types of radiation protection PI could be recognised. These are: occupational exposure of workers, public exposure and management of PI related to radioactive waste. The occupational exposure could be efficiently studied using ISOE database. The dependence of occupational exposure on different types of reactors, e.g. PWR, BWR, are given, analysed and compared.

### Introduction

Regarding [1] altogether 441 nuclear power reactors (NPPs) are in operation worldwide and 32 under construction. Occupational exposure in NPPs is one of the main performance indicators related to safety culture developed in an NPP. It strongly depends on physical characteristics of an NPP as for example:

- type of a nuclear power plant;
- life period of an NPP;
- maintenance and upgrading as for example steam generator replacement;
- refuelling practice of an NPP.

In addition, it also strongly depends on the management of radiation protection issues. As shown in [2] the number of monitored workers all over the world is increasing with time reaching the number around 500 000 of annual monitored workers in the last decade. Worker protection at NPPs is today based on the optimisation principle, so that the doses are as low as reasonably achievable (ALARA).

The Information System on Occupational Exposure (ISOE) which was established in 1992 to provide the forum for radiation protection experts to discuss, promote and co-ordinate the undertakings in the area of worker protection in NPPs [3]. One of the results of ISOE is the ISOE database which includes occupational data from the total 465 reactors in the year 2002, among them 406 operating and 59 in cold shutdown or some stage of decommissioning [4]. The database enables the analysis of exposure regarding different characteristics of the NPPs.

Firstly, the characteristics of the main types of reactors used in commercial purposes are given. Secondly, occupational exposure regarding different types of reactors is compared and analysed.

## Types of reactors

From 1950 many types of reactor have been developed, but today, mainly four types of nuclear power plants widely used:

- light water reactors (LWR);
- heavy water reactors (HWR) as for example Canadian Deuterium Uranium Reactors (CANDU);
- gas cooled reactors (GCR);
- light water cooled graphite moderated reactors (RMBK).

Among them around 75% of all reactors are LWR, either pressurised water reactor (PWR) or boiling water reactors (BWR). The detailed characteristics of the above mentioned are described elsewhere [5].

Figure 1 shows the number of different types of operating reactors in the year 2002 [6]. In this year 321 operating reactors were included in the ISOE database with together 6 014 of years of experiences and among them 192 PWRs with 3 769 years of operating experiences. As shown the majority of operating reactors today use enriched uranium as fuel and light water as a moderator and a coolant.

In Figure 2 the contributions of shutdown reactors of different types participating in the ISOE database in the year 2002 are given. Some of them were not definitively shutdown but they did not operate in the year 2002. In the ISOE database, 43 reactors are included with the mean age of 19.4 years of operation at the time of shutdown. The number of GCRs shutdown is slightly higher than the numbers of other types of reactors shutdown.

**Figure 1. Types of reactors operating during 2002 based on data from** [6]

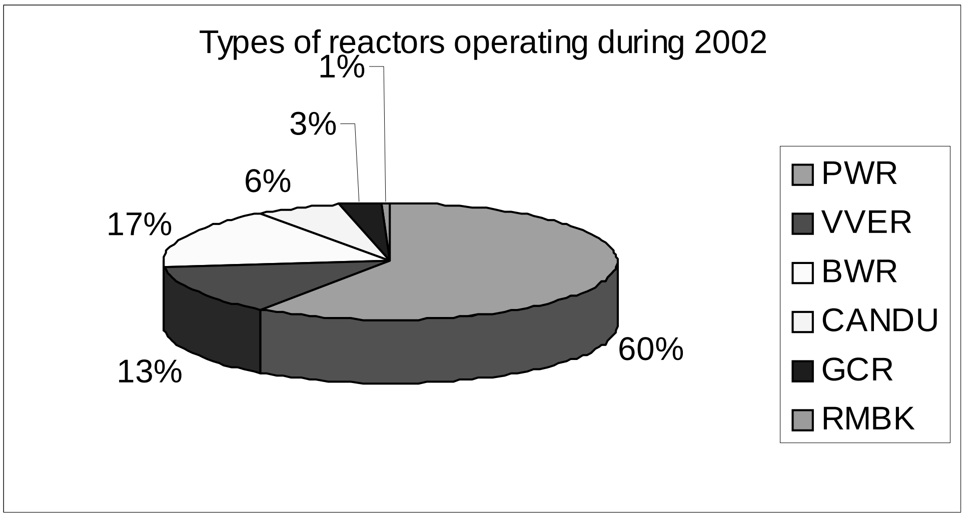
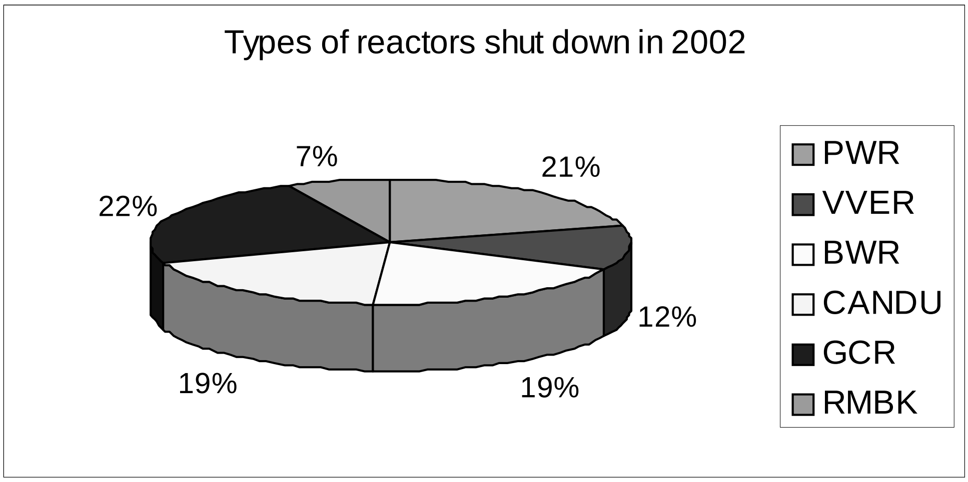



**Figure 2. Types of reactors definitively shutdown as for year 2002 based on data from** [6]




**Occupational exposure in reactors**

Occupational exposure in NPPs strongly depends on the inventory of radioisotopes in:

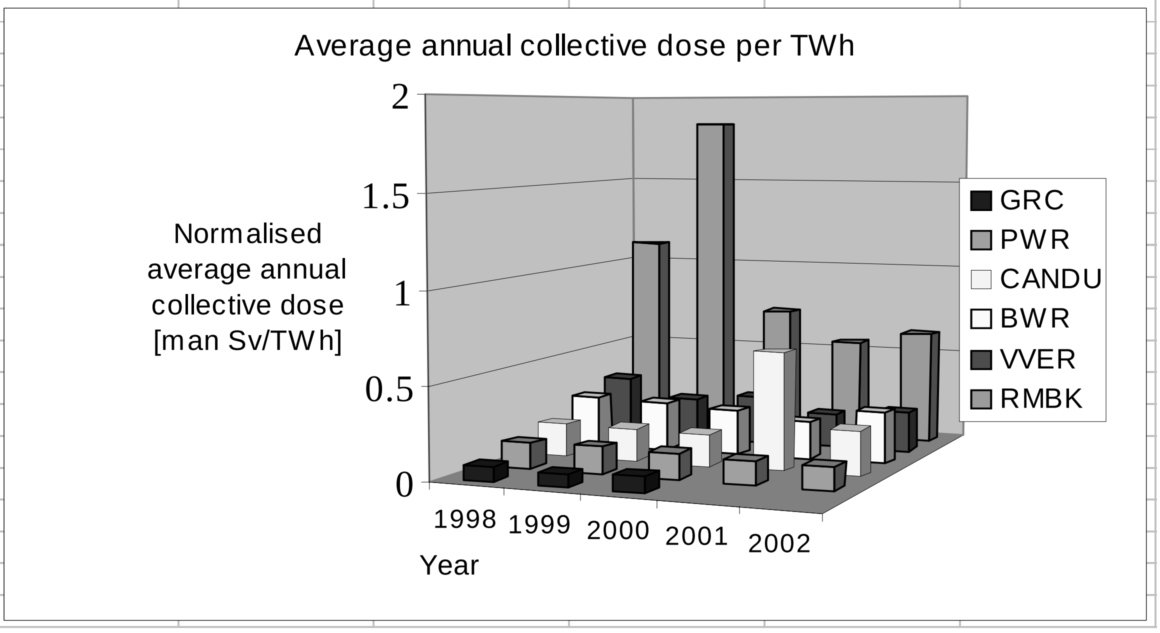
- fuel;
- reactor coolant;
- reactor coolant purification systems and waste–stream processing system;
- radioactive waste.

Moreover also the accumulation of radioisotopes in all additional systems related to the reactor coolant should be carefully studied and monitored in the process of planning the work. The inventory of specific types of reactors can be found in literature [7].

In the year 2002, the average annual collective dose per reactor was 1.06 man Sv. The lowest average collective doses per reactor were obtained in GCR and the highest ones in RMBK with the value 4.40 man Sv. Besides the collective dose as a performance indicator of radiation protection also the normalised average annual collective dose, defined as the average collective dose per reactor per generating electrical energy can be used. Figure 3 shows the normalised average annual collective dose for different types of reactors from the year 1998 to 2002.

**Figure 3. Normalised average annual collective dose for different types of reactors from the year 1998 to 2002**

| PWR | VVER | BWR | CANDU | GCR | LWGR |
|---|---|---|---|---|---|
| 0.140999805 | 0.330650698 | 0.277598335 | 0.180591177 | 0.07905426 | 1.111066844 |
| 0.147354346 | 0.231824515 | 0.268478945 | 0.181429485 | 0.069711766 | 1.840592354 |
| 0.134895523 | 0.268411504 | 0.244558485 | 0.175586845 | 0.086254159 | 0.723542269 |
| 0.12666848 | 0.19085024 | 0.208480125 | 0.636909062 | | 0.552436105 |
| 0.119730928 | 0.223406129 | 0.284080205 | 0.243789835 | | 0.62279027 |

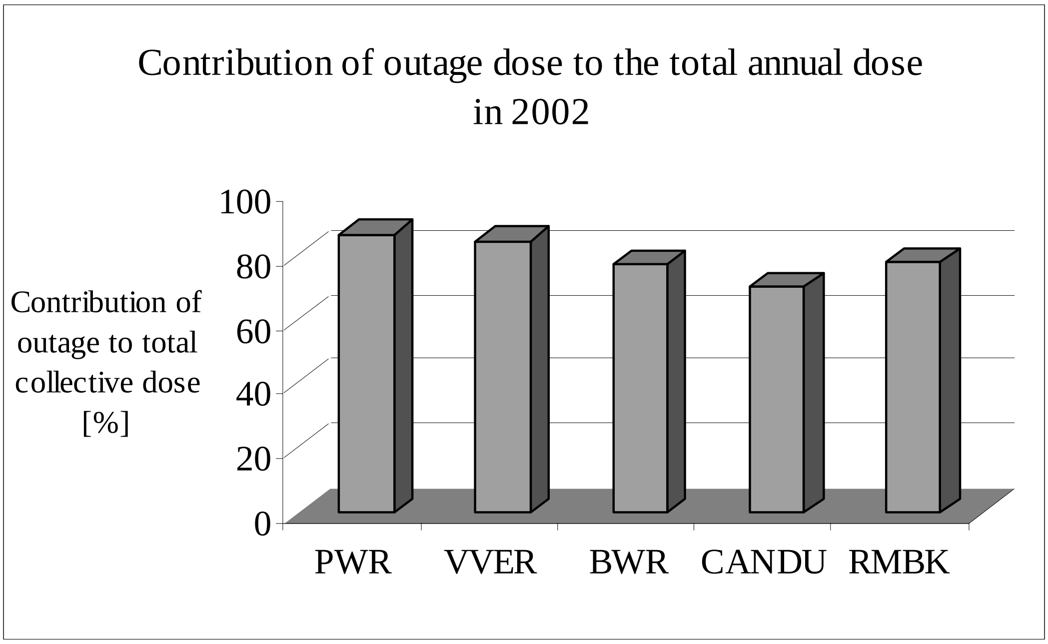



The highest normalised average annual collective dose is regularly observed in RMBK reactors while the lowest one in GCRs in that period. The operation of PWRs is also related to low levels of normalised collective exposure, which is below 0.15 man Sv/TWh in that period.

Detailed analyses of exposure due to specific maintenance tasks or upgrading of NPPs are rare in literature. The analysis of exposure due to the steam generator replacement can be found [8], as well as exposure related to the reactor head replacement. As stated in [9] around 80% of all exposure is usually due to high dose jobs, which can be estimated to represent only about 20% of all jobs in NPPs. A list of typical jobs which are related to high exposures is given for example in [9].

The ISOE database can be used to perform the analysis of specific tasks performed in NPPs. The doses received during an outage are usually much higher that the doses related to normal operation of an NPP. Figure 4 gives the percentage of the outage contribution to the total collective dose for different types of reactor in the year 2002. Values were obtained averages over contributions in all participating countries. As shown the contribution of outage dose to the total dose do not strongly depend on a type of a reactor.

**Figure 4. Average contribution of the outage dose to the total annual dose in 2002 for different types of reactors from in the year 2002**




## Conclusions

The efficient study of workers' exposure regarding six main widely used types of reactors, namely PWR, VVER, BWR, CANDU, GCR and RMBK can be performed using the ISOE database. Occupational exposure performance indicators show that the operation of a GCR leads to the lowest average annual collective dose per energy produced while the highest average annual collective dose per energy exists at RMBK reactors. The outage period is a critical period concerning occupational exposure. The collective dose from that period represents more than 70% of all annual dose and is not a strong function of a reactor type. In PWRs the contribution is around 87% while in CANDU reactors around 70%.

**References**


1. IAEA (2003), *Nuclear Power Reactors in the World*, Vienna.
2. United Nation Scientific Committee on Effects of Atomic Radiation (2000), *Sources and Effects of Ionising Radiation, UNSCEAR 2000 Report to the General Assembly*, United Nations, New York.
3. NEA (2002), *ISOE – Information System on Occupational Exposure, Ten Years of Experiences*, OECD, Paris.
4. NEA (2004), *Twelfth ISOE Annual Report 2002*, OECD, Paris.
5. Knief, R.A. (1981), *Nuclear Energy Technology, Theory and Practice of Commercial Nuclear Power Plants*, Hemisphere Publishing Corporation, Washington, New York, London, 1981.
6. NEA (2004), *ISOE Database, Authority Database*, Version 4.2. (Rev. 63), OECD, Paris.
7. IAEA (1997), *Generic Assessment Procedures for Determining Protective Actions during a Reactor Accident*, Vienna.
8. IAEA (2002), *Optimisation of Radiation Protection in the Control of Occupational Exposure*, Safety Reports Series No. 21, Vienna.
9. NEA (1997), *Work Management in the Nuclear Power Industry*, OECD, Paris.

## MULTIFACTORIAL ANALYSIS OF OCCUPATIONAL OUTAGE DOSES DISPERSION IN THE FRENCH NPPS 1998-2002

**C. Lefaure, L. D'Ascenzo, P. Crouail**
CEPN, France

**G. Cordier, J. Lebeau, A. Rocher, G. Machicoane**
EDF, France

### Introduction

Since 1992, many improvements have been achieved at EDF in terms of reducing occupational radiological exposures. Many efforts have been done in several areas, but without knowing what are the factors that have most influenced these results. It would therefore be of interest now to know more about that in order to be able to decide where to invest with an optimal efficiency to further improve occupational radiological protection.

### Objectives and methods

At the request of EDF, CEPN has performed an in depth statistical analysis in order to reveal the respective influence of factors explaining outage doses dispersion in France, and therefore to determine which factors are still potential levers for continuing improvement of occupational radiological protection. The study has been first carried out among more than 137 outages corresponding to the period 1998-2000; the period has then been expanded with two extra years 2001 and 2002, which has allowed studying 226 outages. Eleven qualitative variables and 39 quantitative variables were selected to take into account reactors design features (10 variables), characteristics of the operation of the plants (15 variables), and characteristics of the shutdown (15 variables), and of the outages themselves (13 variables) (see Annexes 1 and 2). Many of these variables are similar to variables requested in the so-called level 2 questionnaire of ISOE (materials and components, primary water chemistry and contamination levels, hot spots…).

Most of the results from the first study have been confirmed in the second one; therefore, we will here present the results on the five years period, proposing an analysis of the evolution between the three first years and the recent period only in a few cases when the results of the first period have not been confirmed. The main differences between the first step and the second, is that the impact of some variables that was not significant became significant, as the size of the samples had increased.

The study has been performed for the outages sample as a whole as well as for each type of French Pressurised Water Reactors and for each major type of outages (simple refuelling outage (ASR), short maintenance outage (VP) and long decennial outage (VD).

To achieving the previous objectives, CEPN has performed a statistical study in two steps: an analysis of the qualitative variables influence on outage doses, followed by an analysis of correlations between doses and each quantitative variable. All calculations have been performed using STATGRAPHICS Plus software.

**Results**

- Analysis of the influence of the qualitative variables on outage doses dispersion.

This study confirmed the very important impact of the design on the collective dose level during outages. Among the 5 years analysed, doses from the 900 MWe units are 30% higher than those of the 1 300 MWe units. The same gap may be observed for the simple refuelling outages (ASR) as well as for the short maintenance outages (VP). It is even more important between the two types of reactors (40%) for the ten years outages (VD).

The study confirms a statistically significant impact of SG tubes material and fabrication as well as of SG channel head electropolishing on outage dose.

It is interesting to see that there is no relationship between the use of MOX fuel and outage doses, which let consider that there is no significant impact of the MOX fuel on the source term.

One may have expected that, on the 900 MWe units, the type of reactor boron and water make up system might have had an impact on the level of the doses: actually, there are two systems for the cover of the tank corresponding to that system, one with air under the cover and one with nitrogen under the cover; it was therefore suspected that any bore injection in the system with air, will have introduced oxygen into the primary circuit, leading to oxidation and contamination. This is not verified, or at least there is no significant difference with those reactors with nitrogen under that tank cover.

| **Qualitative variables** | Total sample | Outage type | | | 900 MWe | | | | 1 300 MWe | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | ASR | VP | VD | All | ASR | VP | VD | All | ASR | VP | VD |
| Reactor type | + | + | + | + | | | | | | | | |
| Outage type | + | | | | + | | | | + | | | |
| Presence of hot spots | + | + | + | | + | + | + | + | – | | – | |
| Use of MOX fuel | – | – | – | – | – | – | – | – | | | | |
| Large tasks | + | – | + | – | + | | + | – | + | | + | – |
| SG tube material | + | + | + | + | + | + | + | – | + | – | – | |
| SG tubes manufacturer | + | + | + | – | + | – | – | – | – | – | + | – |
| Electropolishing SG channel head | + | – | + | | | | | | – | – | + | |
| SG type | + | + | + | + | + | – | – | – | | | | |
| Reactor boron and water make-up system | + | + | + | + | + | – | – | | | | | |

+ Significant relationships since the first step of the study; + new significant relationships with increased size of the sample in the second step. The blocks in black are those where any analysis should be avoided (*a priori* not meaningful). The blocks with – are those where no relationship is significant.

- Analysis of the influence of the quantitative variables on outage doses dispersion.

Moreover, to study relationships between dose and quantitative variables, doses from units with hot spots have been normalised by dividing them with a 1.3 factor, as a 30% extra dose has been pointed out as a significant average impact resulting from these hot spots on outage doses.

Furthermore, due to the influence of the reactor type and outage type on doses, most analysis have been performed successively on the total sample, samples by type of outages, samples by type of reactors, samples crossing the previous variables (when enough data exist in the sub-samples). This has allowed comparing more homogeneous reactors as regards these variables.

It is not surprising to find that the most important variable, which presents, by far, the highest correlation coefficients (more than 0.7), not only for the sample as a whole, but for each sub-sample by type of reactors (900 MWe or 1 300 MWe) and even by type of outage for each type of reactor, is the time spent in controlled area. Having that in mind, it should be very important to create and follow up new variables that might influence the dispersion of these "time spent", and on which it may be expected to have an influence: percentage of mishaps, reworks, fortuitous works, work management, frequency and justification of controls… It is obvious that some doses (and money!) may still be saved in reducing the time spent in controlled area, as the impact of that variable remains significant even within homogeneous outages for the same type of reactor to explain differences in terms of doses. This comforts the expectations from ISOE a few years ago when issuing the book on "work management".

Another conclusion from that part of the study is that the time spent in the controlled area is a good estimate of the "actual exposed workload", contrarily to the length of the outage in terms of days, which was often referred to in the past as a good variable. Therefore, it seems very important in the future that the information on the time spent in the controlled area will be well followed both at the outage as a whole and at the task levels to allow performing useful benchmarking nationally and internationally.

That variable is therefore the first variable explaining outage doses dispersion. It is not possible to demonstrate that any other quantitative variable (but for a few very specific exceptions that will be pointed out later on) taken alone has a significant impact on the dose dispersion

One can then suppose that all other impacts are hidden by the influence of the time spent in controlled area. To remove that influence, a very efficient solution is the one proposed by our Spanish colleagues at Tarragona [1]: the elaboration of a ratio "outage dose divided by the time spent in controlled area" the so-called "dose index". That outage dose index might be considered as the average dose rate "used" during the outage as a whole.

All statistical correlations have then been performed between the remaining variables and the outage dose index to reveal the influence of other operating and outage variables. At that stage the impact of other variables becomes significant. The first of them is the radiological state of primary circuit (so called "indice de tranche" in the French plants). That variable is the average of a few dose rates measured on the cold and hot legs of the primary pipes in the SG containments with very precise conditions (positions, time after shutdown, circuit configuration…) in all French plants, following recommendations from the SMRP Programme of EPRI. That variable is nevertheless not as powerful as the time spent in the controlled area for explaining the outage doses dispersion, but one has to have in mind that radiological state of the primary circuit, by definition does not take into account the contamination of auxiliary circuits where (or in the vicinity of which) many works are performed. In that context the radiological state of primary circuit only becomes a very powerful variable (correlation coefficient higher than 0.7) for the ten years outages (VD) where much more work is performed on the primary circuit than during the other outages (on may even notice that for the ten years outages the radiological state of primary circuit is significant not only to explain the dispersion of the dose index but also to explain the dispersion of the outage doses themselves).

It is important to point out that even if cobalt 60 is very important during the first cycles, it remains a significant pollutant impacting dose rates and doses with ageing of the units as seen in the study for the 900 MWe units. It confirms the importance of stellite reduction programmes for that type of reactors as well as the importance of reducing $CO_{59}$ content in the steels at both design and modification stages.

When analysing other radio-elements potentially contributing to dose through gamma emissions, some other become also significant contributors for the 1 300 MWe units as cobalt is less important. It re-emphasises the importance of a circuit purification strategy not only focused on cobalt, but also adapting the chemical specifications according to the type of pollutant (Ag, Sb…). EDF has developed such a strategy with all its partners (EDF headquarter departments, CEA, FRAMATOME). It will be now implemented.

The time spent in the controlled area by the health physicists appear also as an important variable for the 900 MWe units. The highest the time, the lowest are the doses with quite good correlations whatever the type of outage (ASR, VP or VD). The impact of this variable to explain the dispersion of doses has been even strengthened with the data from two the last two years. One may wonder why the same impact may not be observed on the 1 300 MWe. An analysis of the data shows that the time spent in the controlled area by the health physicists is much more important (66% more) on the 1 300 MWe units than on the 900 MWe units (75% more for the small outages ASR), with a smaller dispersion (the minimum values are between two and four time lower for the 900 MWe units depending on the year; the standard deviations are quite similar in absolute values, i.e. lower in relative values). Therefore the discrepancy between the two types of reactors is not surprising and one can then expect in the future that a global increase of the time spent in the controlled area by the health physicists in the 900 MWe units, particularly for those where that time is the lowest, will facilitate a reduction of the outage doses.

The time spent by the managers was also introduced as a potential variable to explain the dispersion of outage doses. Unfortunately the correlations are quite good but positive, of course it does not mean that the presence of the mangers is not useful, but clearly it just demonstrate that the longer are the outages, the highest the time spent in the controlled area by the managers and the highest the outage doses.

In a first time (three first years), the time spent in the controlled area by the individuals in charge of decontamination has appeared important for the 1 300 MWe units. This is not any more the case when taking two more years. This does not mean that decontamination is not useful for reducing doses, but it just corresponds to an important reduction of the dispersion of the time spent during the different outages (by more than a factor 3).

As well as the impact of the qualitative variable “type of reactor boron and water make up system”, some quantitative variables representatives of the number of rapid and important “load modifications” during the cycle, were introduced to check if the corresponding water movements may have introduced a significant oxidation and therefore an increase of the dose rates and doses. Correlations are no more significant with these variables than the impact of the qualitative variable. This comforts the conclusions of EMMEC measurements performed by the French Atomic Energy Commission that these load modifications cannot be significantly related to the outage doses.

Another variable has been introduced for checking the impact of the corrosion: the number of days during the cycle with “pH” lower than 6.9. It was expected that the higher that number, the higher the acidity and the corrosion in the primary circuit, the higher the dose rates nearby the primary circuit. What has been first observed is a difference of behaviour between the 900 MWe and 1 300 MWe, as there is nearly two times more days for the 1 300 MWe: 20 days ± 12 instead of

12 days ± 8 (but one has to have in mind that the fuel cycle is 18 months for the 1 300 MWe while it is one year for the 900 MWe). Secondly, in most situations no significant correlation with either the dose index or the radiological state of primary circuit have been observed, but for the 1 300 MWe during short outages only during the first period of the analysis (1998-2000) with both the radiological state of primary circuit and the dose index. The main characteristics of that sub-sample is that there has been an important increase of the average number of days between 1999 and 2000 (from 12 days to 23) with an increase of the dispersion, while it remains stable from 2000 to 2002. The example of this variable (and some of the previous) shows both: – that, even if some phenomenon's are obvious (impact of acidity on corrosion…) the discrepancies between the units on the selected indicators are often not enough important to show significantly that they play a role for explaining the dispersion in terms of outage doses or dose index; – furthermore that it is not obvious to find good indicators.

**Conclusion**

The results presented here are just examples of what can be done with such a study both in France and at the international level with ISOE 2 data when fulfilled; they may be considered as preliminary results, they must be confirmed and validated.

Of course, some variables such as Zinc injection have not been studied here, as they were not relevant in France. Some others have not been kept such as biological shielding tons installed during the outage, as the information was not available in the French plants.

However that study has pointed out the interest of mixing qualitative and quantitative variables, the interest of mixing descriptive statistical analysis with the analysis of correlations between dose dispersion and other variables dispersions as well as there evolution with time.

*Annex 1.*
**QUALITATIVE VARIABLES**

| Variables | Modalities |
|---|---|
| Design variables | |
| Type of reactor. | 900 MWe, 1 300 MWe. |
| Generation. | Three generations for the 900 MWe and 2 for the 1 300 MWe. |
| Reactor boron and water make up system. | Three modalities (with air, with nitrogen, floating cover). |
| Steam Generator Type. | 51A, 51B, 51BI, 51M, 4722, 6819. |
| SG tube material. | MA600, TT600, TT690. |
| Electropolishing SG channel head. | Yes, No. |
| SG tubes manufacturer. | Vallourec, Sandvick, Westinghouse. |
| Operation variable | |
| MOX fuel used. | Yes, No. |
| Shutdown variables | |
| Outage type. | Simple Refuelling outage (ASR), Partial checking outage (VP), Ten years outage (VD). |
| Large tasks. | SGR, VHCR, … |
| Presence of hot spots. | Yes, No. |

The first two variables correspond to the definition of the sister unit groups of the ISOE System.

## *Annex 2.*
## QUANTITATIVE VARIABLES

| Design variables | |
|---|---|
| SG cobalt content (in ppm) | |
| Number of standard fuel clusters | |
| Number of fuel clusters with cladding | |
| **Operation variables** | |
| Average pH at cycle starting | |
| Number of days with pH < 6.9 | |
| Average $H_2$ content (in ml/kg-TPN) | |
| Cycle length (in equivalent fuel power days) | |
| Number of abnormal operating situations (three set of modalities) | |
| Fuel enrichment rate | |
| Number of MOX assemblies | |
| Age | |
| Number of days with reduced power | |
| **Variables at outage start** | |
| Maximum values at the oxygenation peak (in MBq/t): | |
| - $\gamma_{total}$, | - Ag110m, |
| - Co58, | - Sb124, |
| - Co60, | - Sb122, |
| - Cr51, | - rapport $\gamma_{total}$/Co58 |
| Purification length since oxygenation peak (in hours) | |
| Maximum values at the primary pumps stop (in MBq/t): | |
| - Co58, | - Ag110m |
| - Co60, | - Sb124 |
| - Cr51, | - Sb122 |
| **Variables during shutdown** | |
| Managers collective dose | |
| Health Physicists collective dose | |
| Collective dose for decontamination | |
| Total man-hours in controlled area | |
| Total managers man-hours in controlled area | |
| Total health physicists man-hours in controlled area | |
| Total man-hours in controlled area for decontamination | |
| Number of extra days when shutdown prolonged | |
| Dose rate index nearby primary circuit (in 1E-2mSv/h) | |

# OCCUPATIONAL RADIATION PROTECTION AT SWEDISH NUCLEAR POWER PLANTS: VIEWS ON PRESENT STATUS AND FUTURE CHALLENGES

**I. Lund, S. Erixon, T. Godås, P Hofvander, L. Malmqvist,**
**I. Thimgren,** and **H. Ölander Gür**
Swedish Radiation Protection Authority, Sweden

## Abstract

The occupational radiation doses at Swedish NPPs have decreased with roughly a factor of two from the beginning of the 1990s until today. The average collective dose during the last five years is 10 manSv for eleven operating reactors. During the same period, the average annual individual dose to the personnel has decreased from 3-4 mSv/year to about 2 mSv/year. In this presentation, the measures taken to improve the radiological conditions at the NPPs are briefly reviewed and the present status is described. The expectations for the future are outlined. The SSI summarises past experiences and the prerequisites for preserving good radiation protection conditions by the following catch words: *Competence, Experience Feedback, Preventive Measures,* and *Long-term Planning*.

## The Swedish nuclear programme

Sweden has eleven operating nuclear power reactors while two reactors have closed operations. Three of the reactors are pressurised water reactors, PWRs, delivered by Westinghouse Monitor AB and put into commercial power production in the period 1975-1983. Eight reactors are boiling water reactors, BWRs. They were delivered by ASEA Atom AB and started their commercial power production during the period 1972 (Oskarshamn 1) to 1985 (Oskarshamn 3 and Forsmark 3).

The two closed reactors, the Ågesta reactor and Barsebäck 1, were shutdown in 1974 and 1999, respectively. In Table 1, the main data for the Swedish nuclear power programme is summarised.

The electric power in Sweden is almost entirely produced in hydro and nuclear power stations. In the period 1990-2001, on average, nuclear power accounted for 46% of the total electric power production in Sweden.

## External factors

Good radiation protection conditions are principally the result of good awareness of, and commitment to, radiation protection at the nuclear facility. External factors, however, influence the radiation protection issues.

**Table 1. Main data for the Swedish Nuclear Power Programme**

| | Power (th) MW | Type | Operator | Commercial operation |
|---|---|---|---|---|
| **Ågesta** | 105 | PHWR | AB Atomenergi/Vattenfall | 1964-74 |
| **Barsebäck 1** | 1 800 | BWR | Barsebäck Kraft AB | 1975-99 |
| **Barsebäck 2** | 1 800 | BWR | Barsebäck Kraft AB | 1977- |
| **Forsmark 1** | 2 928 | BWR | Forsmarks Kraftgrupp AB | 1980- |
| **Forsmark 2** | 2 928 | BWR | Forsmarks Kraftgrupp AB | 1981- |
| **Forsmark 3** | 3 300 | BWR | Forsmarks Kraftgrupp AB | 1985- |
| **Oskarshamn 1** | 1 375 | BWR | OKG Aktiebolag | 1972- |
| **Oskarshamn 2** | 1 800 | BWR | OKG Aktiebolag | 1975- |
| **Oskarshamn 3** | 3 300 | BWR | OKG Aktiebolag | 1985- |
| **Ringhals 1** | 2 500 | BWR | Ringhals AB | 1976- |
| **Ringhals 2** | 2 660 | PWR | Ringhals AB | 1975- |
| **Ringhals 3** | 2 783 | PWR | Ringhals AB | 1981- |
| **Ringhals 4** | 2 783 | PWR | Ringhals AB | 1983- |

***International organisations***

The work practices and the radiation protection philosophy applied, as well as rules and regulations, in Sweden, are framed and formulated in interplay with views, recommendations and rulings from international organisations.

The International Commission on Radiological Protection, the ICRP, has a leading role in defining and recommending norms and principles within the radiation protection area. The three fundamental principles of the ICRP's protection philosophy: *Justification, Optimisation* and *Dose Limitation* are widely accepted and used. Sweden became a Member of the European Union in 1995 and has implemented the fundamental Directive 96/29/EURATOM *Council Directive of 13 May 1996 laying down basic safety standards for the health and protection of the general public and workers against the dangers of ionising radiation* into Swedish legislation.

The International Atomic Energy Agency, IAEA, develops and sets standards, which are worldwide recognised and accepted – both in the area of safety and radiation protection. Apart from "*Expert missions*" and "*benchmarking activities*", personnel from Swedish authorities and Swedish nuclear industry are involved in the work related to the Convention on Nuclear Safety (Sweden's second national report under the Convention of Nuclear Safety, Ds 2001:41 *Ministry of the Environment*).

The Nuclear Energy Agency of the OECD, NEA, assists its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe use of nuclear energy. Sweden has actively been involved in the NEA work, covering areas like radioactive waste management, radiation protection philosophy, decommissioning issues, and environmental radiological protection.

The World Association of Nuclear Operators, WANO, has as its mission "to maximise the safety and reliability of the operation of nuclear power plants by exchanging information and encouraging communication, comparison and emulation amongst its members". Some Swedish NPPs have been subject to benchmarking activities co-ordinated through WANO.

***Authority requirements***

The regulations of the Swedish Nuclear Power Inspectorate, SKI, and the SSI have changed during the past years. Since the beginning of the 90s the safety requirements have increased and the SKI has formulated new regulations about non-destructive testing, safety barriers and staff competence. The SKI is presently in the process of reviewing and updating its main safety regulations, which were issued in 1998. The SSI has, during the last ten years, implemented the common European radiation protection legislation, formulated in binding EC directives, into the Swedish regulations.

*Deregulation of the electricity market*

The Swedish electricity market was deregulated in 1996 when open competition was introduced in trade and production of electricity. The grid system is still regulated and controlled. The company Svenska Kraftnät owns the national grid and has the role of system operator. The trade is performed at Nord Pool – The Nordic Power Exchange in Oslo, Norway.

The deregulation has led to changes in the financial situation for the power producers, rationalisation and new organisational structures. One way of decreasing costs has been to review the investment plans. All power producers do not, however, compete on the same terms since a special excise duty is imposed on nuclear power.

*Environmental protection issues*

Under the last ten years, the interest in environmental issues has continued to increase. For the operators of nuclear power plants, and in the general debate, releases of radioactive substances has come into focus and received a greater attention than earlier. The discussions and the work performed in relation with the environmental issues has also led to the application of a changed protection philosophy: The releases should be reduced if it is possible with reasonable technical efforts even if the resulting radiation doses to the most exposed persons are small. The concept of *best available technique*, BAT, is applied.

The issue of risk transfer between individuals and groups of individuals is a complex issue. When the protection of the environment (eco system) should be considered, the SSI finds it important that new routines for optimisation are established and applied – enabling appropriate attention to occupational exposure in the optimisation process.

**Past radiation protection conditions**

In the beginning of the 90s the SSI observed a trend of increasing occupational exposure at the Swedish Boiling Water Reactors. The main reasons for this were extended reconstruction work at the reactors and increase in non-destructive testing leading to more work in the controlled areas of the NPPs. It was also noted that radiation levels in water-filled systems at the power plats were still increasing and had not levelled as expected. The SSI and the nuclear industry then implemented measures in order to change the situation and to reduce incurred and projected radiation doses.

**Actions to improve conditions**

In revised regulations adopted in 1994 the SSI required each utility to prepare special programmes with the aim to reduce occupational doses and radiation levels (ALARA programmes). The SSI also required extended education and training programmes in radiation protection, addressed particularly to foremen and team leaders working at the NPPs. Another important regulatory measure taken by the SSI was the introduction of the dose limit of 100 mSv in five consecutive years (max. 20 mSv as average over five years) in addition to the annual dose limit of 50 mSv. SSI has actively supported research and development projects for understanding, modelling and reducing radiation doses and radiation levels at NPPs.

The nuclear industry introduced programmes with the aim to avoid further increase of radiation levels at the NPPs. One important step is to reduce the amount of cobalt entering the reactor, by exchange of components or passivating surfaces containing Stellite. During the past ten years, at maintenance and modernisation programmes performed at the plants, piping and vaults have been exchanged. This has decreased the cobalt inventory but also introduced materials less susceptible to corrosion, which affects the need for maintenance and time intervals for in-service inspections. Other important steps for reducing the build-up of radiation levels have been to control the water chemistry in reactor circuits (e.g. Zn-injection, Ni/Fe-ratio) and optimise water flows. Great care was taken in improving different start and shutdown procedures (pH-values, temperature conditions) and other operations, which can lead to unnecessary spread of contamination in the plant. Special filters to catch debris and unwanted pollutants were sometimes installed. The operators have selected to use chemical decontamination more frequently in connection with complex work in high radiation areas. In a few BWRs, lowering the moisture content of the reactor steam successfully reduced the radiation levels at the turbine side.

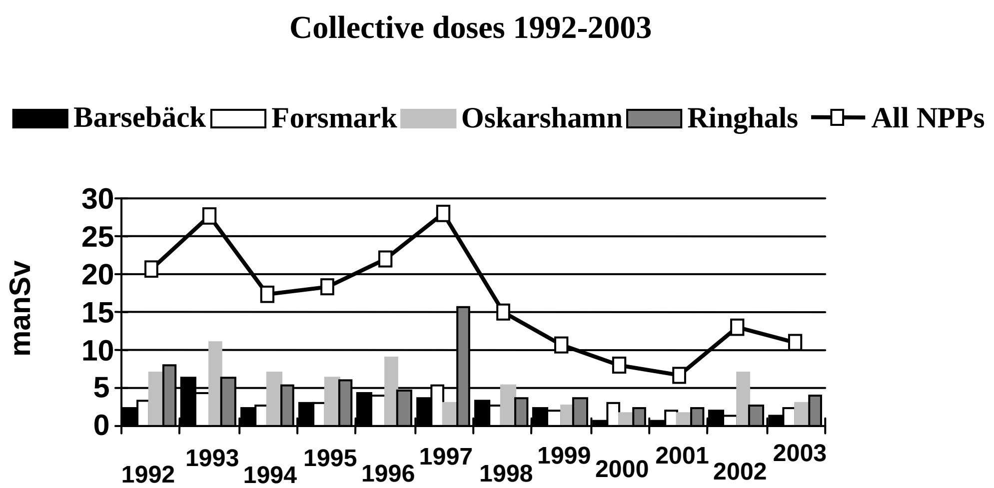
Among the administrative measures, apart from improved education plans and the use of mock-up facilities, changes have been introduced in the planning of outages and projects. Healths physics staff is involved early in the planning and pre-planning phase in order to introduce radiation protection aspects at the design stage (e.g. space, material choice, work procedures). Written procedures for fuel damage management and policy for when to interrupt production in connection with serious damages have been developed by the operators. Efforts to minimise fuel damages have led to internal rules in order to restrain debris and filings to enter the primary reactor systems.

Both international and national systems for feedback and exchange of experiences (e.g. ISOE, WANO, INPO) have been utilised for improving work procedures. An important and still on-going part of the work is to improve the co-operation with external contractors in the field of radiation protection and ALARA planning.

Several research programmes were ordered by the SSI and the nuclear industry in order to improve knowledge and techniques for reducing radiation levels. A few examples are:

- model development for activity build-up adopting theories for surface complexes and diffusion in oxide layers;
- radiological effects of hydrogen water chemistry and Nobel metal chemistry addition;
- KEMOX 2000 – Kinetics of oxide layers;
- project DORIS – Dose reduction in Swedish BWRs;
- the fuel failures in Oskarshamn 2 1988 – An evaluation of the radiological effects during ten years of operation.

**Figure 1. Collective doses at Swedish NPPs during 1992-2003**



## Present status

After a decade, the positive results of the combined actions from the SSI and the Swedish nuclear industry can be observed. Occupational doses have decreased and the radiological environment in the reactors has improved. Figure 1 shows the development of collective radiation doses at Swedish NPPs during 1992-2003.[1] As can be seen in the figure the collective dose has decreased from about 2 manSv in the beginning of the 90s to about 1 manSv in the last five years. It is the view of SSI that the occupational doses, today and during the passed years, would have been higher if no counteractions had been introduced in the beginning of the 90s. The average individual dose has in the same time interval decreased from 3-4 mSv/year to about 2.5 mSv.

The increase in radiation levels (apart from re-oxidation of contaminated surface layers) was generally stopped and in some plants lower levels were achieved due to the efforts to reduce the production and distribution of Cobalt 60. Low contamination levels and improved work procedures are also reflected in the low number of reported intakes of radionuclides. The number of reported intakes (leading to a committed effective dose larger than 0.25 mSv) is presently 1-2 per year.

## The Future – catchwords and outlook

### *Competence*

A basic condition for good performance in the radiation protection area is that the staffs is familiar with the risk of ionising radiation, actions to prevail unnecessary exposures and the meaning of good practice. The changed education programmes, earlier introduced at the nuclear power plants, had a significant influence on the workforce competence as well as the general commitment to radiation protection items. SSI continues to underline the importance of the plant management commitment to radiation safety issues and the use of preventive measures to decrease dose rates and doses.

SSI recently inspected the radiation protection education programmes at the NPPs. SSI has the view that in the next few years it is possible to maintain the level of competence and secure necessary educations and training. In order to secure good long-term radiation protection conditions at the Swedish nuclear power plants it is, however, important that national support of natural sciences and nuclear technology can be sustained. The SSI is today using a large fraction of its research budget to support critical competence areas at the universities (professorial chairs, postgraduate appointments) such as radiobiology, radiation medicine, radio physics and radioecology.

### *Experience feedback*

An important task in developing radiation protection is the use of channels for exchange of information and feedback of good/bad practices. The operators use both national and international

1. Since the number of operating reactor units is 12 (11 after 1999) it is possible to scale the y-axis with a factor of ten to get a good estimate of the average collective dose per reactor and year.

information systems. The ISOE (Information System of Occupational Exposure), organised by OECD/NEA and IAEA, is used for exchange on dose statistics, information and technical issues. Another important system is INES (International Nuclear Event Scale) through which occurring accidents/incidents are classified and communicated to the media and the public.

The SSI stresses the need for openness and transparency within the nuclear industry. This will lead to early reporting on poor situations, the use of good practices, and improvements. In the present Swedish nuclear industry safety culture, feedback from both incidents and good practice are reported, collected and analysed. Communication, both internal and external, should be encouraged as a natural part of the daily work. It is important to retain this openness also in the future and it must therefore be a central issue in the organisational work. A key resource is every individual worker who must feel responsibility and commitment to report and inform within and outside of the organisation without the risk of punishments or repressive actions.

***Preventive measures***

Good radiation protection conditions are achieved by proactive actions and preventive measures. The necessary work starts with the source-term, i.e. to prevent distribution and build-up of radioactive nuclides on the reactor system surfaces. The Swedish nuclear power industry has improved the radiation levels at the stations by improving the reactor water chemistry and by selecting new, more appropriate materials in valves and piping. Development of new methods and new equipment for non-destructive testing are other examples of preventive measures. It should also be recognised that radiation protection issues are now considered in the early planning stage of projects and maintenance work.

The SSI is led to believe, based on the development of new safety requirements and on the expressed wish to increase power production in existing power plants, that more refurbishment work will take place at the Swedish reactors. The SSI therefore identifies as one of its principle future tasks to, in the dialogue with the plant operators, ensure that radiation protection issues are adequately addressed in these processes. Sufficient resources should be allocated and radiation protection issues have a reasonable priority, in order to maintain and possibly improve radiological protection conditions.

***Long-term planning***

For the industry to invest, technically as well as in human resources, in increased safety and bettered radiation protection, there is a need for long-term perspectives and long-term planning. It is important to see beyond the short-term perspective and try to foresee and meet the future needs. This is true when competence, research, technical development as well as economical investments are addressed.

On the political agenda, discussions are presently held between the Swedish government and the nuclear power industry on the future use of nuclear industry and how the politically decided phase-out of nuclear energy should be performed. If such an agreement is reached, it could perhaps improve on the existing situation in the sense that uncertainties are removed and improved planning of maintenance, repair and modernisation of the nuclear power plants can be performed. It is the experience of the SSI that long-term views and planning in advance improve radiation protection conditions.

### Final conclusion

It is the view of the SSI that essential efforts to improve the radiation protection conditions at the Swedish nuclear power plants have been made. The radiation protection conditions are good, which is a result of long-term efforts on reducing radiation levels, improving work procedures as well as increasing the knowledge of, and the commitment to, radiation protection issues at the staff level. Some of the most important aspects that led to the present good radiological situation can be summarised by the following catchwords: *Competence*, *Experience Feedback, Preventive Measures* and *Long-term Planning*.

# SESSION III

**MANAGEMENT OF TRITIUM EXPOSURES FOR PROFESSIONALLY EXPOSED WORKERS AT CERNAVODA 1 NPP**

**V. Simionov**
Cernavoda NPP, Romania

## Abstract

Operating experience to date of CANDU reactors has indicated that the major contributor to the internal dose of professionally exposed workers is the tritiated heavy water (DTO).

CANDU reactors are both moderated and cooled by heavy water ($D_2O$). Tritium is produced in CANDU reactors by neutron reactions with deuterium, boron, and lithium and by ternary fission.

Even small leaks from these systems can produce important contaminations with tritiated water vapours of the air in the reactor building and thus increased individual and collective internal doses.

Professionally exposed workers are subject to a combination of acute and chronic tritium exposure and HTO dosimetry programme at Cernavoda NPP is based on multiple sample results. The routine urine bioassay programme performs the monitoring and dosimetry functions for DTO. A specialised laboratory using liquid scintillation spectrometry methods currently determines tritium activities in urine samples. The frequency of biological samples submission depends on the tritium concentration in the last sample.

Dose assignments resulting from routinely measured weekly and monthly urinary levels of tritium oxide are based on the method of linear interpolation unless it is known that there has been no exposure between samples (vacation).

All information about these doses is stored into a dedicated electronic database and used to make periodical reports and to ensure that the legal and administrative individual and annual limits are not exceeded.

A chronic unprotected exposure to small tritium dose rate ($< 50\mu Sv/h$) may lead to internal doses that exceed the intervention level. In case of acute exposure an increased daily water intake combined with a proper medical intervention could reduce the effective half time of tritium 2-3 times.

## Introduction

Situated at 180 km east of Bucharest, Cernavoda Nuclear Power Plant is a CANDU 6 type NPP. CANDU (CANadian Deuterium Uranium) is a Canadian design power reactor, which employs natural uranium as fuel and heavy water as a neutron moderator and as the thermal agent.

The thermal neutron flux in the CANDU reactor, by activation of deuterium, is the major producer of tritium but other nuclear reactions could also produce tritium as listed below.

a) Activation reactions

- ${}^{2}_{1}H + n \rightarrow {}^{3}_{1}H$
- ${}^{6}_{3}Li + n \rightarrow {}^{3}_{1}H + {}^{4}_{2}He$
- ${}^{10}_{5}B + n \rightarrow {}^{3}_{1}H + 2\,{}^{4}_{2}He$

b) Ternary Fission.

c) Reconversion of $^{3}$He from $^{3}$H Decay.

Very small amounts of DTO may escape from moderator and heat-transport systems of CANDU reactors during maintenance and normal operation.

Even small leaks from these systems can produce important contaminations with tritiated water vapours of the air in the reactor building and thus increased individual and collective internal doses. That why the ventilation systems were carefully designed to control water vapour concentration in radiological areas and special dryers remove moisture from the air in order to maintain the tritium doses well below the limits.

Despite the protection measures operating experience to date of CANDU reactors has indicated that the major contributor to the internal dose of professionally exposed workers is the tritiated heavy water (DTO) which is present chronically at many work locations.

**Tritium characteristics**

Exposure to an atmosphere contaminated by tritiated water results in intake of that substance both by inhalation and by absorption through the intact skin, in a ratio assumed to be 2 to 1.

Vapours of triturated water are considered to be of SR-2 absorption class that means the tritiated water is instantaneously absorbed into body fluids and uniformly distributed among all the soft tissues and is eliminated with a nominal half time of 10 days. In addition a very small fraction is incorporated in non-exchangeable form and eliminated with a much longer half time.

Tritium (H-3) is a pure beta emitter, with an average energy of beta radiation of 0.0057 MeV. Its presence in the body can be detected by measuring the urine samples using the liquid scintillation counting and it presents no detection problems.

**Internal dosimetry for DTO**

The principal objectives of individual monitoring for intakes of radionuclides are:

- to obtain an assessment of the committed effective dose;
- to contribute to the control of operation and the design of the plant;
- in the case of accidental exposure, to provide valuable information for the initiation and support of any appropriate health surveillance and treatment.

Professionally exposed workers are subject to a combination of acute and chronic tritium exposure and DTO dosimetry programme at Cernavoda NPP is based on multiple sample results. Body DTO concentration is integrated over time and multiplied by the dose rate per unit concentration factor as in relation.

$$E = 5.8 \cdot 10^{-2} \sum_{i=0}^{k-1} [(C_{i+1} + C_i)/2] \cdot (t_{i+1} - t_i)$$

where E is the effective dose in mSv. The urine concentrations $C_i$ are given in MBq/L, and the time is expressed in days. Tritium doses are registered into personal records with a registration level of 0.17 mSv.

The committed dose (mSv) associated with a $^3$H concentration C (MBq/L) in case of an acute intake is computed as follows:

$$E_{(50)} = 0.84 \cdot C$$

where the dose factor 0.84 was computed by using tritium physical characteristics, anatomic and metabolic data for Reference Man [Popescu, Chitu, 2001].

***Bioassay for intakes of DTO***

Bioassay monitoring for internal dosimetry of DTO is relatively simple involving the sampling of a single void urine sample. The method consists of mixing 1 mL of urine sample with 10 mL of scintillation cocktail. (Packard ULTIMA GOLD ™ ). This mixture is well shacked for 10 minutes to ensure the homogeneity of the sample and then measured by the liquid scintillation spectrometer CANBERRA-PACKARD TR/LL – 2550.

A monthly frequency of bioassay submission is used at Cernavoda NPP for professionally exposed workers who are infrequently exposed or exposed to low tritium levels (urine concentration remains below 100 kBq/L).

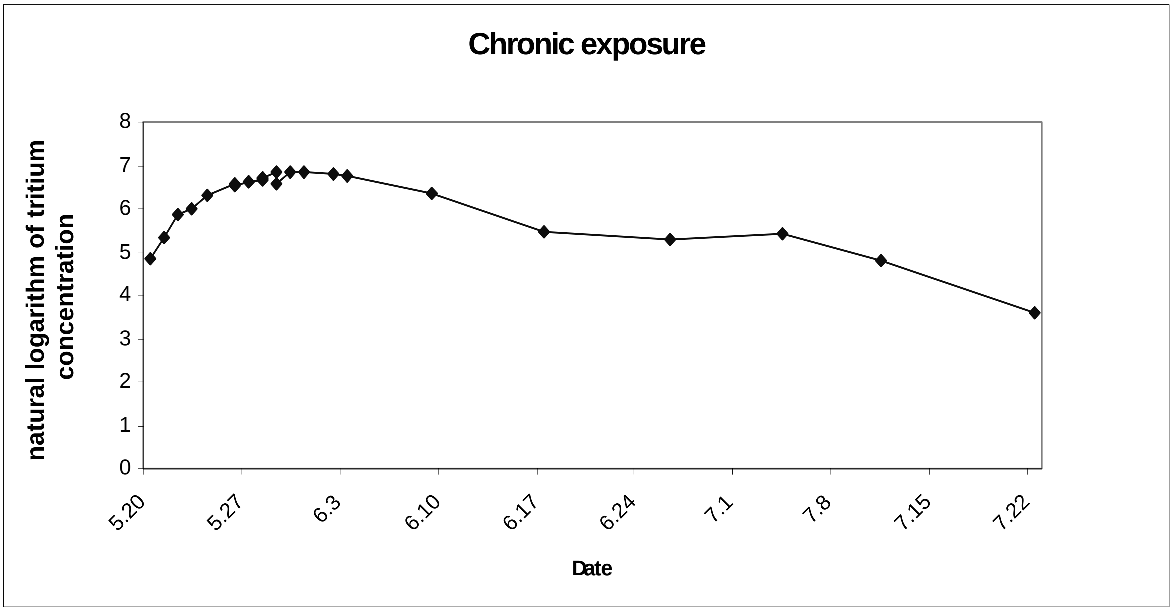
If the urine tritium concentration is greater than 100 kBq/L weekly sampling will be required.

When concentration exceeds 1 MBq/L, the investigation level, daily sample submission is required.

In case of acute exposures, which significantly exceed chronic levels, the most important error in dosimetry arises from the estimation of the time of intake. Therefore special monitoring is required when planned exposures to DTO are foreseen, the worker should submit additional samples before and after the task completion. When working conditions are unexpectedly changing and could produce abnormal exposures to DTO, all the personnel involved will submit additional samples.

Dose assignments resulting from routinely measured weekly and monthly urinary levels of tritium oxide are based on the method of linear interpolation unless it is known that there has been no exposure between samples (vacation).

**Figure 1. Chronic exposure**



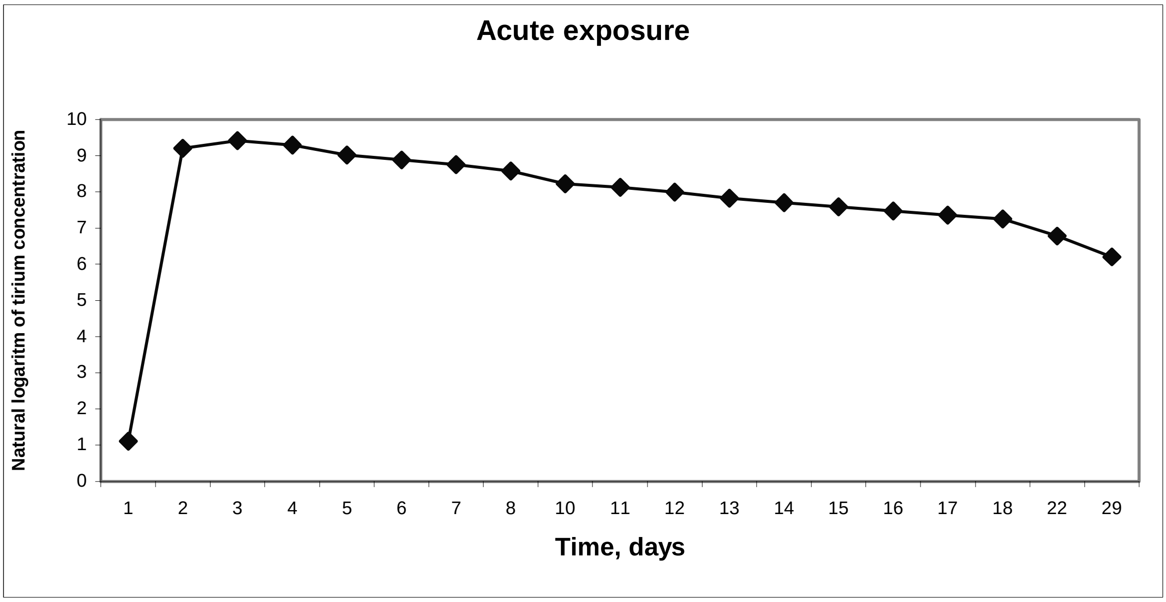

A chronic unprotected exposure to small tritium dose rate (< 50μSv/h) may lead to internal doses that exceed the intervention level, as can be seen in Figure 1.

In case of acute exposure dose-mitigating actions are recommended by the Occupational Medicine Specialist in consultation with Dosimetry Programme responsible. The primary treatment for reducing internal dose from a tritiated water uptake is to accelerate the turnover of body water. This can be done by substantially increasing the fluid intake rate of an individual through oral or intra-venous means, and/or using diuretics. Cernavoda NPP experience intakes indicating that a sustained drinking regime gave a clearance half-time of about 5-6 days compared with a 10 day normal clearance half-time.

Figure 2 illustrates tritium dynamics in urine following unusual DTO incorporation based on daily measurements. Tritium clearance was accelerated with diuretics under physicians' surveillance, which resulted in low tritium effective half-times, 5.4 days. This value is obviously smaller than the mean value of 10 days which is conservativelly used in dosimetric calculations.

During the investigation period this worker was not allowed to enter in tritium contaminated areas until the tritium concentration in urine had decrease below 1 MBq/L.

**Figure 2. Tritium dynamics in urine**




All information about these doses is stored into a dedicated electronic database and used to make periodical reports and to ensure that the legal and administrative individual and annual limits are not exceeded.

## Conclusions

Tritium is an important contributor to the internal exposure of radiation workers in Cernavoda NPP. Collective dose for professionally exposed workers reached a value of 818.28 man mSv in 2003 and internal doses raised from 1.9% in 1996 to about 40% in 2002.

Table I presents the internal dose distribution by dose interval due to tritium intake between 1999 and 2003. As can be seen most of the results were below the recording level, the majority of recordable doses were less then 1 mSv.

**Table I. Internal dose distribution (mSv) by dose interval 1999-2003**

| Year | 0.0 | >0.0 <1.0 | 1.0-5.0 | 5.0-10.0 | 10.0-15.0 | 15.0-20.0 | Over 20.0 |
|---|---|---|---|---|---|---|---|
| 1999 | 1 353 | 236 | 23 | 0 | 0 | 0 | 0 |
| 2000 | 1 399 | 243 | 32 | 0 | 0 | 0 | 0 |
| 2001 | 1 419 | 327 | 37 | 0 | 0 | 0 | 0 |
| 2002 | 1 571 | 343 | 57 | 1 | 0 | 0 | 0 |
| 2003 | 1 580 | 505 | 83 | 0 | 0 | 0 | 0 |

The actual levels of internal doses due to tritium exposures reveal the effectiveness of implementation of the Radiation Safety Policies and Principles established by the management of the Cernavoda NPP, based on the ALARA principles.

**References**


1. ICRP Publication 60, 1990, *Recommendations of the International Commission on Radiological Protection*, Pergamon Press.
2. ICRP Publication 30, 1979, *Part I, Limits for intakes of radionuclides by workers*, page 65, Pergamon Press.
3. ICRP Publication 68, 1998, *Dose coefficients for intakes of radionuclides by workers*, page 22, Pergamon Press.
4. *Personnel Dosimetry Program for Cernavoda NPP*, Reference document RD-01364-RP6/Rev.2, Chap. 3.II-2.1.4, Tritium – Dosimetric calculations.
5. IAEA (1996), Safety Series No. 115, *International Basic Safety Standards for protection against Ionising Radiation and for the safety of Radiation Sources*, Table II-IX, Vienna, page 274.

# EVIDOS: OPTIMISATION OF INDIVIDUAL MONITORING IN MIXED NEUTRON/PHOTON FIELDS AT WORKPLACES OF THE NUCLEAR FUEL CYCLE

**M. Luszik-Bhadra[1], M. Boschung[3], M. Coeck[4], G. Curzio[5], D. Derdau[8], F. d'Errico[5], A. Fiechtner[3], J.-E. Kyllönen[6], V. Lacoste[2], Th. Lahaye, L. Lindborg[6], C. Molinos[7], H. Muller[2], M. Reginatto[1], H. Schuhmacher[1], R. Tanner[7], F. Vanhavere[4]**
[1]Physikalisch-Technische Bundesanstalt, Germany
[2]Institut de Radioprotection et de Sûreté Nucléaire, France
[3]Paul Scherrer Institut, Switzerland
[4]Studiecentrum voor Kernenergie-Centre d'étude nucléaire, Belgium
[5]Dipartimento di Ingegneria Meccanica, Nucleare e della Produzione, Italy
[6]Swedish Radiation Protection Authority, Sweden
[7]National Radiological Protection Board, United Kingdom
[8]Kernkraftwerk Krümmel GmbH, Germany

## Introduction

Within its 5th Framework Programme, the EC is funding the project EVIDOS ("Evaluation of Individual Dosimetry in Mixed Neutron and Photon Radiation Fields"). The aim of this project is the optimisation of individual monitoring at workplaces of the nuclear fuel cycle with special regard to neutrons. Various dosemeters for mixed field application – passive and new electronic devices – are tested in selected workplace fields in nuclear installations in Europe. The fields are characterised using a series of spectrometers that provide the energy distribution of neutron fluence (Bonner spheres) and newly developed devices that provide the energy and directional distribution of the neutron fluence. Results from the first measurement campaign, carried out in simulated workplace fields (IRSN, Cadarache, France), and those of a second measurement campaign, carried out at workplaces at a boiling water reactor and at a storage cask with used fuel elements (Kernkraftwerk Krümmel, Germany), are described.

## Selection of workplace fields

A first important task of the project was the selection of workplace fields. There were two main aims: one was to select places with significant contributions of neutrons to the personal dose equivalent but which differ with respect to neutron energy and direction, neutron to photon dose ratio and environmental conditions in terms of noise, temperature, vibrations and electromagnetic interference. The second aim was to involve the radiation protection officers at the facilities and to promote discussions. Some of the first project meetings, either with all members of the project or within smaller task groups, were held at the envisaged facilities. The workplaces were visited and their suitability discussed. The facilities considered are listed in Table 1.

The campaign C0 was carried out at the simulated workplace fields at Cadarache. These fields are particularly attractive for performance tests of electronic dosemeters which may not present a perfect response in quasi-mono-energetic neutron fields but respond with sufficient accuracy in practical fields with broader energy distributions. Secondly, these fields have been characterised with extensive MCNP calculations of energy and directional distributions of neutron fluence(1) and are thus well suited for testing of the performance of the new spectrometers.

Electronic neutron personal dosemeters are especially needed for operational dosimetry in nuclear power plants and there is also an upcoming need for neutron monitoring around casks that contain spent fuel. Places which are routinely visited for control measurements at the Krümmel Nuclear Power Plant were chosen for measurement campaign C1: a position in the control room underneath the reactor, another position near the top of the reactor and two places close to a cask containing spent fuel.

**Table 1. List of sites for the measurements**

| | Facility | Place | Measurement period |
|---|---|---|---|
| C0 | Canel and Sigma simulated workplace neutron fields | Cadarache, France | October/November 2002 |
| C1 | BWR Krümmel | Krümmel, Germany | April 2003 |
| C2 | Venus Research Reactor, SCK•CEN and Nuclear Fuel Facility Belgonucléaire | Mol, Belgium | June 2003 |
| C3 | PWR Ringhalsverket | Ringhals, Sweden | planned for fall 2004 |
| C4 | Fuel processing plant/Magnox Reactor | Sellafield, United Kingdom | planned for spring 2005 |

In Mol, the measurements for the C2 campaign were performed at two facilities. At the VENUS Research Reactor one location was chosen where personnel need to read a gauge next to the reactor and another in the control room. At the Belgonucléaire fuel processing plant, four positions were chosen: bare plutonium rods, plutonium-rods in a rack with and without shielding and inside a stock room.

The measurement campaign C3 is foreseen to take place at the PWR Ringhalsverket. This reactor is of a different type (PWR) from the Krümmel reactor (BWR). Positions with extremely severe environmental conditions are selected. This allows the performance of dosemeters and spectrometers to be tested under harsh conditions. In addition, these fields have been extensively investigated by spectrometric methods several years before, thereby providing additional information and comparisons.

Finally, campaign C4 is intended to take place at a fuel processing plant at Sellafield in the United Kingdom.

## Dosemeters and spectrometers used

The dosemeters used in the EVIDOS project are listed in Table 2. Recent publications concerning their performance are given in the references in Table 2 and some recent overviews.[11 13]

The IRSN Bonner sphere spectrometer is used for reference spectrometry. It consists of an $^3$He filled proportional counter and 12 polyethylene spheres. The five smallest spheres are used bare and also with a cadmium shield. This system is well characterised by calculations and measurements at monoenergetic fields.[14] The measurements were performed at the PTB and NPL standard laboratories and at the SIGMA IRSN thermal neutron facility.[15]

The simultaneous measurement of neutron fluence as a function of energy and direction is performed with two novel instruments: one based on Si-diodes mounted on a stationary polyethylene sphere and one using a SDD-spectrometer inside a rotatable collimator.

The directional spectrometer based on silicon detectors[16] consists of six detector capsules, each containing a stack of 4 silicon detectors, mounted onto the surface of a 30 cm diameter polyethylene sphere and electronics to amplify and record the pulse height spectra of all detectors. The response function of this device has been determined for a series of directions using measurements in quasi-mono-energetic neutron fields and MCNP calculations for neutrons in the energy range from thermal up to 15 MeV, and using measurements for photons in the energy region from 80 keV to 6 MeV. The pulse height spectra measured in workplace fields are analysed using unfolding codes with respect to energy and direction, for both neutrons and photons.[16,17]

One of the codes (MIEKE) works without explicit pre-information on energy and directional distribution, the other one (MAXED) takes pre-information into account: with respect to the neutron energy, primarily the spectra measured at the same places by Bonner spheres were taken for pre-information. For the direction, initial estimates were derived from the raw detector readings of the directional spectrometer itself.

**Table 2. Short description of the devices used in the EVIDOS project**
The status [commercial (c) or prototype (p)] is given in the last column

| Name of device | Short description | |
|---|---|---|
| BAE SYSTEMS | novel area monitor for H*(10) and $H_p(10,\alpha)$ measurements[2] | p |
| Berthold LB 6411 | moderator type area monitor | c |
| Harwell N91 | moderator type area monitor | c |
| Sievert instrument | low pressure proportional counters (one of tissue-equivalent plastics and one of graphite) evaluated according to the variance/covariance technique[3] | p |
| Studsvik 2200s | moderator type area monitor | c |
| Wendi-II | moderator type area monitor with tungsten loaded moderator[4] | c |
| Aloka PDM-313 | **electronic neutron dosemeter with 1 silicon detector** | c |
| BTI-PND | fast neutron bubble detector[5] | c |
| DISN | differential reading of two ionisation chambers, based on Direct Ion Storage two types, each type with and without boron plastic shieldings[6] | p |
| DOS-2002 | electronic photon/neutron dosemeter with 1 silicon detector[7] | p |
| HpSLAB | superheated drop detector inside a slab phantom[8] | p |
| PADC (CR-39) | track etch detector (chemical + electrochemical etching)[9] | c |
| PND+BDT | combination of fast and thermal neutron bubble detector | c |
| Saphydose-n | electronic neutron dosemeter using a segmented silicon diode[10] | c |
| Siemens EPD N | electronic photon/thermal neutron dosemeter with 3 silicon detectors | c |
| Siemens EPD N2 | electronic photon/neutron dosemeter with 3 silicon detectors | c |

The directional spectrometer with superheated drop detectors[18] uses a “telescope design” with a single detector at the centre of a 30 cm diameter moderating-sphere of nylon-6. The system views a narrow solid angle of about 1/6 steradians since the hydrogenous sphere effectively attenuates laterally incident neutrons, thus providing a strong angular dependence of the response. By changing the temperature of the superheated drop detector from 25°C to 55°C, a series of responses with threshold behaviour is obtained as a function of neutron energy. The response functions have been determined experimentally using quasi-monoenergetic neutrons for frontal incidence and using MCNP calculations for higher angles of incidence. First analyses of the energy and direction of neutrons were performed using the MAXED unfolding code.[17,18]

## Results in simulated workplace fields

First results of dosemeter readings obtained in the Cadarache simulated workplace fields were presented at the $9^{th}$ Symposium on Neutron Dosimetry in Delft.[19] Two fields were discussed: the thermal field SIGMA,[15] with 43% of the ambient dose equivalent arising from thermal neutrons, and the CANEL[20] field with a broad energy distribution showing a maximum contribution to the dose equivalent in the range of a few hundred keV and additional significant contributions of thermal and intermediate energy neutrons. Reference values were obtained from Bonner sphere measurements and MCNP calculations.

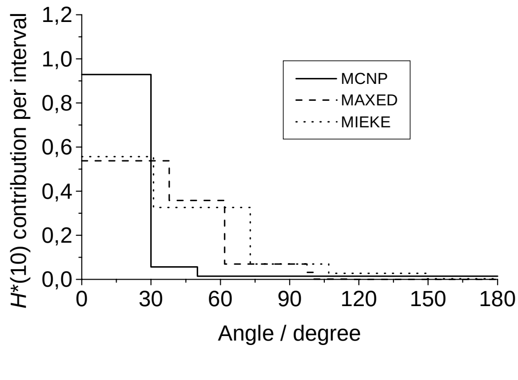
All personal dosemeters were attached to ISO phantoms for irradiation. For normal incidence, the dosemeters Aloka PDM-313 and Siemens EPD N2 showed over-readings by more than a factor of two compared to the reference values, while others (Saphydose-n, DOS-2002, PADC track detectors from NRPB and devices based on superheated drop detectors like HpSLAB and BTI PND) showed deviations less than 30%. DIS-N dosemeters were used with and without boron plastic shielding. Without shielding over-responses of up to a factor of 20 were recorded, while with shielding responses between 0.5 and 1.1 were obtained. The Siemens EPD N, a dosemeter which is intended only for the measurement of thermal and intermediate energy neutrons, responded well to the thermal part of the spectrum at SIGMA but showed under-readings by more than a factor of two at CANEL.

The readings of the ambient dose equivalent devices deviated by less than 30% from the reference values, except for one case: BAE systems at SIGMA showed a response value of 0.59.

Both directional spectrometers were used at SIGMA and CANEL. Due to time consuming variations of the temperature of the superheated drop detectors and the small effective opening of the "telescope device" measurements were performed only in one direction (frontal) at SIGMA and two directions (frontal and 30°) at CANEL. The analysis using unfolding codes is still in progress.

The results of the directional spectrometer with silicon detectors were analysed and first results published.[16] Figure 1 shows the directional distribution obtained at CANEL. $H^*(10)$ contributions per angular interval, derived from MCNP calculations, decrease sharply above 30°, whereas the results of the directional spectrometer show significant contributions up to 60°. This shows the limited angular resolution of the experimental device. Each of the six capsules mounted onto the surface of the sphere chiefly extracts information on neutron fluence impinging on a solid angle cone corresponding to roughly one sixth of the total solid angle. The angle integrated fluences and ambient dose equivalent showed agreement with the reference values within 20%.

**Figure 1. H*(10) contribution per angular interval as calculated by MCNP for the CANEL field and derived from the directional spectrometer using the unfolding codes MIEKE and MAXED**



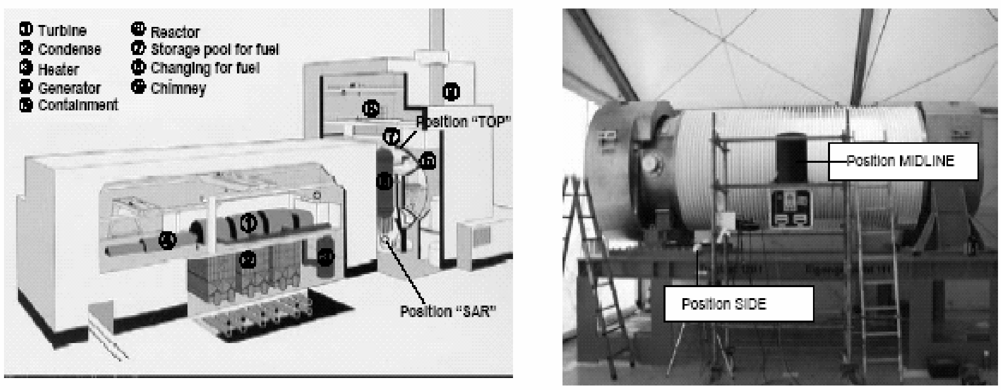
## Results at Krümmel

The measurements were performed at two positions inside the boiling water reactor and at two positions near an NTL11 cask with spent fuel (see Figure 2 and Table 3).

**Table 3. Measuring positions at Krümmel**

| Position | Distance 1 | Distance 2 | Height above floor | Front direction |
|---|---|---|---|---|
| KKK TOP, 40 m level | to door: 0.8 m | to wall: 0.68 m | 1.50 m | towards reactor |
| KKK SAR | centre of room | centre of room | 1.20 m | towards lock |
| Cask midline | centre of cask | to cask: 1.0 m | 2.65 m | towards cask |
| Cask side | 10th ring, left side | to cask: 1.35 m | 1.50 m | towards cask |

**Figure 2. Measuring positions at Krümmel**




The results obtained with the Bonner sphere spectrometer for the energy distribution of neutrons are shown in Figure 3 together with angle integrated spectra obtained from analysis of the directional spectrometer with silicon detectors. The spectra at the cask are quite hard, with a main fluence contribution at a few hundred keV, while the spectra at the reactor contain a considerable amount of thermal and intermediate energy neutrons.

Reference values for $\dot{H}^*(10)$ were obtained by multiplying the spectra by fluence-to-ambient dose equivalent conversion coefficients (see Table 4). The results of the ambient dose equivalent meters agree with the reference values $\dot{H}^*(10)$, within 30%, except for Wendi-II (1.75 at KKK SAR) and for BAE systems (0.66 at KKK TOP). More detailed information on results of ambient dose equivalent meters measured at "cask midline" and at SAR is given in Reference 19.

$\dot{H}^*(10)$ values obtained by the directional spectrometer with silicon detectors agree within 30% with the reference values in case of the MAXED unfolding (see also Figure 3) and showed deviations up to 50% for unfolding without pre-information (MIEKE). Preliminary estimates for $\dot{H}_p(10)$ were derived using the following approach: The spectra, obtained for different directions from measurements with the directional spectrometer, were multiplied with fluence-to-personal dose conversion coefficients [new ones calculated also for backward directions(21)] and fluence-to-ambient dose equivalent conversion coefficients, and ratios $\dot{H}_p(10)/\dot{H}^*(10)$ were calculated. Values derived from the directional spectrometer using both unfolding codes (MAXED and MIEKE) are given in

Table 4. Low values for the ratio $\dot{H}_p(10)/\dot{H}^*(10)$ were obtained in the field with almost isotropic distribution of neutron fluence, at the SAR. Preliminary estimates for $\dot{H}_p(10)$ were obtained by multiplying the $\dot{H}^*(10)$ reference values obtained using the Bonner spheres with values of $\dot{H}_p(10)/\dot{H}^*(10)$ obtained using the angular spectrometer and MAXED and MIEKE unfolding (mean value). Estimate of $\dot{H}_p(10)$ for the front direction are given in Table 4. The last column in Table 4 contains the ratio of neutron and photon contributions of ambient dose equivalent (latter measured by a FHT 191N ionisation chamber).

**Figure 3. Spectral neutron fluence per logarithmic bin-width as a function of energy at two workplaces investigated during C1 (Krümmel) using Bonner spheres (dotted line)[19] and the directional spectrometer with silicon detectors (full line), MAXED unfolding**

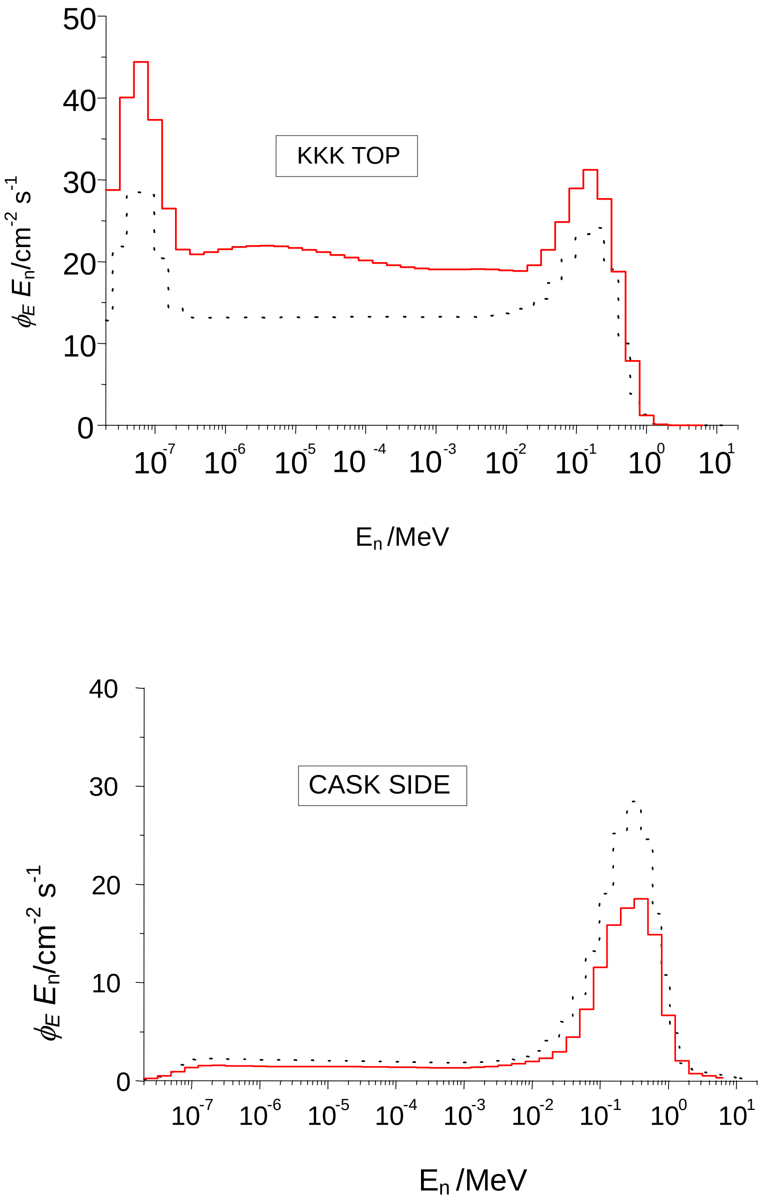


**Table 4. $\dot{H}$*(10) reference values and estimates of $\dot{H}_p$(10) (latter for front direction) for the measurement positions at Krümmel. Preliminary values of uncertainties are given.**

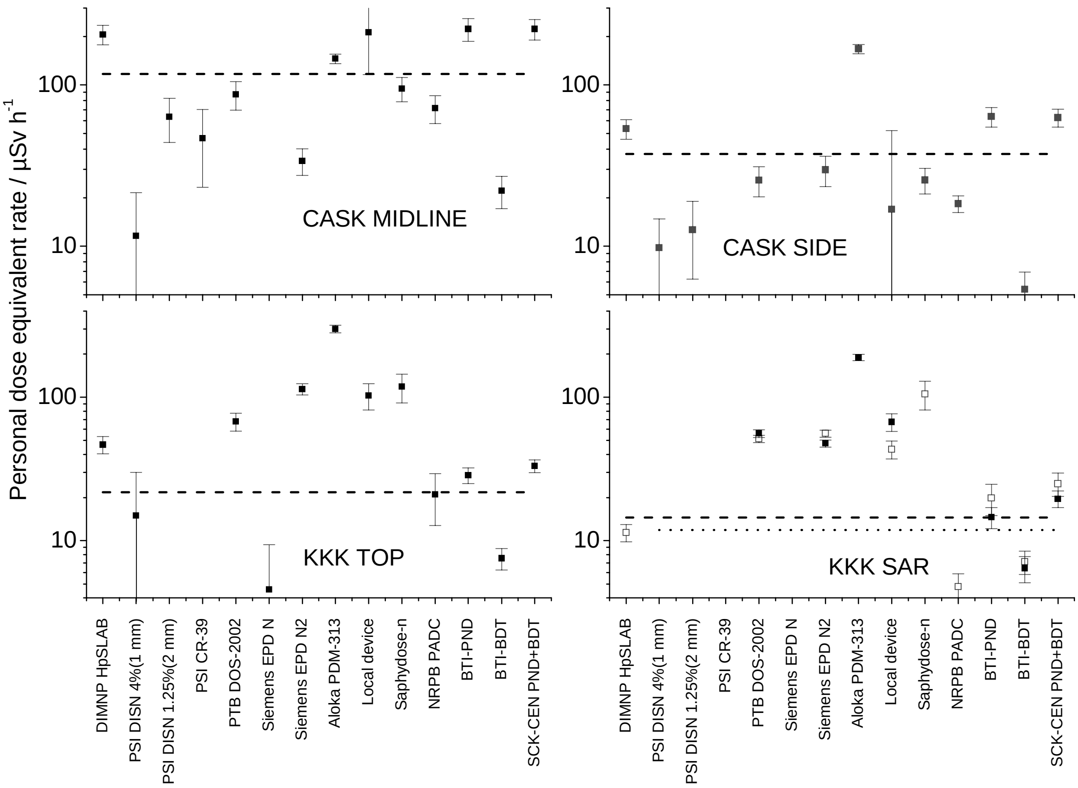
| Position | $\dot{H}^*(10)$/μSv h$^{-1}$ | $\dot{H}_p(10)/\dot{H}^*(10)$ MAXED, front | $\dot{H}_p(10)/\dot{H}^*(10)$ MIEKE, front | $\dot{H}_p(10)$/μSv h$^{-1}$ front | $\dot{H}^*_n(10)/\dot{H}^*_\gamma(10)$ |
|---|---|---|---|---|---|
| KKK TOP | 39.0 ± 5.8 | 0.52 | 0.60 | 21.8 ± 6.5 | 4.20 |
| KKK SAR | 46.8 ± 7.0 | 0.33 | 0.29 | 14.5 ± 4.4 | 0.24 |
| Cask midline | 152 ± 23 | 0.69 | 0.85 | 117 ± 35 | |
| Cask side | 54.6 ± 8.2 | 0.52 | 0.85 | 37.4 ± 11.2 | 40 |

The personal dosemeters were irradiated on ISO phantoms in the positions “cask side”, “cask midline” and “KKK TOP” for about 2 hours and inside the SAR under nitrogen atmosphere for 38 hours. In the latter case, the track dosemeters were sealed in air in order to avoid track fading due to oxygen deficiency. The readings of the personal dosemeters attached on the front side are shown in Figure 4. In all cases also dosemeters were attached on the back side of the phantom. The readings were mostly zero or close to the detection limit of the dosemeters and are not shown in Figure 4 besides those measured at the SAR, where the radiation field was much more isotropic.

The estimated values of $\dot{H}_p(10)$ are indicated by dashed/dotted lines. In both reactor fields some of the personal dosemeters seem to over-respond $\dot{H}_p(10)$ by more than a factor of two. This is probably due to over-responses of these dosemeters for thermal and intermediate energy neutrons. Please note that the uncertainties of the estimated values of $\dot{H}_p(10)$ are still of the order of 30% (one standard deviation).

**Figure 4. Personal dose equivalent rates $\dot{H}_p(10)$ of dosemeters at Krümmel**

The dashed lines indicate estimated values of $\dot{H}_p(10)$. The local device was a TLD albedo dosemeter. The full/open symbols indicate values measured at the front side/back side of the phantom ($\dot{H}_p(10)$ value dotted for back side).




## Outlook

To achieve the aim of the project a consistent description and understanding of all measurements and results is necessary. This implies a deeper understanding of the dosemeter responses in workplace fields by multiplying the spectral information by the angle dependent response of the dosemeters. Equally important is the knowledge of energy and direction distribution of neutrons for the investigated fields. Such additional information can be obtained by analysis of the results measured by superheated drop detectors and PADC track detectors mounted in different directions on the sides of the phantom.


### *Acknowledgements*

This research is partly funded by the European Commission under the auspices of the Euratom 5th Framework Programme: Nuclear Energy, 1998-2002, Contract No FIKR-CT-2001-00175.

## References


1. Lacoste, V. and Gressier, V., "Monte-Carlo simulation of the IRSN CANEL/T400 realistic mixed neutron-photon field", *Radiation Protection Dosimetry*. (in press).
2. Bartlett, D.T., Tanner, R.J. and Jones, D.G. (1997), "A New Design of Neutron Dose Equivalent Survey Instrument", *Radiat. Prot. Dosim.* 4, 267-271.
3. Kyllönen, J., Lindborg L. and Samuelsson, G. (2001), "The response of the Sievert instrument in neutron beams up to 180 MeV", *Radiat. Prot. Dosim*. 94, (3), 227-232.
4. Olsher, R.H., Hsu, Hsiao-Hua, Beverding, A., Kleck, H., Casson, W. H., Vasilik, D. G. and Devine, R. T. Wendi (2000), "An improved neutron rem meter", *Health Physics*, 79, 170-181.
5. Vanhavere, F. and d'Errico, F. (2002), "Standardisation of superheated drop and bubble detectors", *Radiat. Prot. Dosim.* 101, (1-4), 284-287.
6. Fiechtner A., Boschung, M. and Wernli, C., "Present status of the personal neutron dosemeter based on direct ion storage", *Radiat. Prot. Dosim*. (in press).
7. Luszik-Bhadra, M., Wendt, W. and Weierganz, M., "The electronic neutron/photon dosemeter PTB DOS-2002", *Radiat. Prot. Dosim.* (in press).
8. D'Errico, F., Apfel, R.E., Curzio, G., Nath, R. (2001), "Electronic Personal Neutron Dosimetry with Superheated Drop Detectors", *Radiat. Prot. Dosim.* 96 (1-3), 261-264.
9. Gilvin, P.J., Bartlett, D.T., Shaw, P.V., Steele, J.D. and Tanner R.J. (2001), "The NRPB PADC neutron personal dosimetry service", *Radiat. Prot. Dosim.* 96 (1-3), 191-195.
10. Lahaye, T., Chau, Q., Ménard, S., Ndontchueng-Moyo, M., Bolognese-Milsztajn, T. and Rannou, A., "Numerical and experimental results of the operational neutron dosemeter 'Saphydose-N' ", *Radiat. Prot. Dosim.* (in press).
11. Luszik-Bhadra, M. "Electronic personal dosemeters: The solution to problems of individual monitoring in mixed neutron/photon fields?", *Radiat. Prot. Dosim.* (in press).
12. D'Errico, F., Luszik-Bhadra, M. and Lahaye, T. (2003), "State of the art of electronic personal dosimeters for neutrons", *Nuclear Instr. and Methods in Physics Research*, A 505 (1-2), 411-414.
13. Luszik-Bhadra, M., Rimpler, A. and Wahl, W. (2003), "Status and prospects of electronic personal dosemeters for neutron/photon radiation fields", *Kerntechnik* 68, 135-139.
14. Lacoste, V., Gressier, V., Pochat, J-L., Fernández, F., Bakali, M. and Bouassoule, T. "Characterisation of Bonner sphere systems at mono-energetic and thermal neutron fields", *Radiat. Prot. Dosim.* (in press).
15. Lacoste, V., Gressier, V., Muller, H. and Lebreton, L., "Characterisation of the IRSN graphite moderated americium-beryllium neutron field", *Radiat. Prot. Dosim.* (in press).
16. Luszik-Bhadra, M., Reginatto, M. and Lacoste, V., "Measurement of energy and directional distribution of neutron and photon fluences at workplace fields", *Radiat. Prot. Dosim.* (in press).
17. Reginatto, M., Luszik-Bhadra, M. and d'Errico, F., "An unfolding method for directional spectrometers", *Radiat. Prot. Dosim.* (in press).
18. D'Errico, F., Giusti, V., Reginatto, M. and Wiegel, B., "A telescope-design directional neutron spectrometer", *Radiat. Prot. Dosim*. (in press).

19. Bolognese-Milsztajn, T., Bartlett, D., Boschung, M., Coeck, M., Curzio, G., d'Errico, F., Fiechtner, F., Giusti, V., Gressier, V., Kyllönen, J., Lacoste, V., Lindborg, L., Luszik-Bhadra, M., Reginatto, M., Schuhmacher, H., Tanner, R., Vanhavere, F. "Individual monitoring in Workplaces with mixed neutron/photon radiation", *Radiat. Prot. Dosim*. (in press).
20. Gressier, V. *et al.* (13 authors), "Characterisation of the IRSN CANEL/T400 facility producing realistic neutron fields for calibration and test purposes", *Radiat. Prot. Dosim.* (in press).
21. Siebert, B. and d'Errico, F. (to be published).

# PASSIVE DOSIMETERS BENCHMARKING

**C. Pauron**
EDF, France

French regulations originally required film to be used as a passive dosimeter. This ruling was changed with the publication of the decision dated 23 March, "stipulating the rules concerning external dosimetry for workers assigned to tasks involving exposure to radiation". This decision lays down certain obligations in terms of results, but does not impose any particular technology.

Moreover, technological advances have made the future of silver film uncertain, for both photographical and dosimetry purposes.

Consequently, in early 2001 the Management at the EDF Group's Nuclear Generation Division (DPN) decided to undertake a study, looking at the feasibility and potential benefits of adopting a new passive dosimeter technology.

**Definitions:**

**TLD:** thermoluminescent dosimeter;

**OSL:** optically stimulated luminescence dosimeter;

**RPL:** radiophotoluminescent dosimeter;

**Study plan and conclusion:**

- 2001:
  State of the art of the various technologies available and experience feedback from foreign nuclear operators.
- 1$^{st}$ half of 2002:
  Comparative study of the behaviour of the different technologies of Film, TLD, OSL and RPL, carried out under laboratory conditions and also in real situations in 3 EDF nuclear plants over a 3-month period. These 4 technologies were selected on the basis that they are used in industry throughout the world.
- Early 2003:
  Survey of international nuclear operators conducted via the ISOE network. The operators surveyed were asked about the technologies they use, any changes they were considering or had already implemented, the reasons for these changes and their appraisal of the technologies concerned.

- 1st half of 2003:
  The DPN Management decided to replace film with OSL in 2 pilot nuclear plants for the period July 2004 to December 2005. If the results of the pilot are positive, film will be replaced with OSL in all plants in 2006.
- September 2003:
  European call for tender published for the period July 2004 to December 2005.

**Comparative trials**

Around 250 badges including Film, TLD, OSL and RPL worn by a specific group of personnel, selected according to exposure conditions.

High exposure:

- maintaining the reactor building (cleaning, decontamination);
- installing and removing nozzle dam inside steam generators water box;
- valve mechanics, insulation technicians and welders working on primary and connected circuits.

Low exposure:

- chemists;
- business managers and foremen;
- RPS (radiation protection service) technician;
- fuel disposal;
- sorting radioactive waste.

Identical badges were tested in extreme conditions:

- exposure to 20 mGy for 1 minute using a gamma radiography projector;
- exposure to 4.4 mGy for 11 minutes by a filter in the process of installation;
- exposure through an X-ray detector machine;
- poor thermal conditions with temperatures reaching 70°C.

The results are consistent, and the deviations observed are in line with standards. The technical performances of the OSL and RPL dosimeters are quite similar, and often superior to those of film and TLD.

**Results of comparative study**

| Characteristics | | Film (KODAK) | TLD (HARSHAW) | OSL | RPL |
|---|---|---|---|---|---|
| Technical aspects | Dose dynamic | – | = | + | + |
| | Dose linearity | – | = | + | ++ |
| | Energy response | – | = | + | ++ |
| | Repetition | – | = | + | + |
| | Batch uniformity | – | + | + | ++ |
| | Influence quantity | – | = | + | ++ |
| Regulatory aspect | Dose storage | ++ | – | ++ | = |
| Economic aspects | Number of suppliers | + | ++ | – | – |
| | Market – Carriers | = | + | + | – |
| | Operating cost | + | – | + | – |
| Summary | | – | = | ++ Homogeneous | + |

– Acceptable = Medium + Good ++ Very good

EDF chose to adopt OSL, since it combines the various benefits of film, TLD and RPL:

- good sensitivity, as with TLD and RPL;
- good linearity with the dose, as with TLD and RPL;
- good energy response, as with TLD and RPL;
- monthly dose re-reading possible, as with film;
- low cost, as with film.

**Summary of the ISOE international survey**

| Country | Plant | Current dosimeter | Change planned Y/N | Planned technology | Reasons |
|---|---|---|---|---|---|
| **United States** | Susquehanna | TLD | Y | OSL | OSL allows "provisional" reading whenever desired |
| | Calvert Cliffs | TLD | Y | OSL | OSL has good accuracy levels and economic benefits |
| | San Onofre | TLD | N | | Any change would involve switching to an electronic dosimeter |
| | Commanche Peak | TLD | N | | No change, given the investment made in TLD |
| **Canada** | Gentilly | TLD | N | | |
| **United Kingdom** | Sizewell | Film | Y | Electronic dosimeter | Legally recognised as dosimeter |
| **Sweden** | Oskarshamn | TLD | N | | Film has been definitively rejected due to its problems and poor detection limit |
| | Ringhals | TLD | N | | |
| **Germany** | Brokdorf | Film | N | | Any change would involve switching to an electronic dosimeter |
| | Neckarwestheim | RPL | N | | |
| **Belgium** | Doel + Thiange | Film and TLD | N | | |
| **Czech Republic** | Dukovany | Film | N | | Any change would involve switching to an electronic dosimeter |
| **Bulgaria** | Kozloduy | TLD | N | | |
| **South Africa** | Koeberg | TLD | Y | OSL | OSL is cheap. Excellent directional response. Re-reading possible. Very good neutron response compared with TLD. Independent laboratory. |
| **Brazil** | Angra | Film | Y | TLD | Accuracy and cost. TLD is already used, and a certification procedure is under way to have TLD confirmed as legally recognised dosimeter. |

| Country | Plant | Current dosimeter | Change planned Y/N | Planned technology | Reasons |
|---|---|---|---|---|---|
| **Japan** | Fukushima<br>+ Kashiwasaki | TLD | N | | TLD replaced film in 2000. Disadvantages of TLD: no re-reading, regular calibrations required. |
| | Tomari<br>+ Onagawa<br>+ Shika<br>+ Takahama<br>+ Shimane<br>+ Ikata<br>+ Gonkai | RPL | N | | RPL replaced film in 2001. |
| | Hamaoka +Tokai<br>+Tsuraga | Electronic dosimeter | N | | Replaced film or TLD. |

### References


1. *State of the art of various technologies*: study conducted by Mr. Javaraly Fazileabasse of EDF Research and Development – Process Performance Optimisation Department (R&D/OPP).
2. Comparative study conducted by Mr Javaraly Fazileabasse of EDF R&D/OPP. TLDs were loaned and interpreted by COGEMA and OSLs by the company LANDAUER. The CEA provided the RPLs. Trials were organised in real situations by Mr Charles Pauron of the Risk Prevention Group, which is part of the EDF/DPN Operational Installations Support Centre, (CAPE-GPR). Mr Pauron worked in conjunction with the Risk Prevention Services at the Gravelines, Penly and Tricastin nuclear plants. The trial protocol was submitted for approval by the French Institute for Nuclear Safety and Radiation protection (IRSN). A copy of the report was also sent to the institute. EDF/R&D/OPP will be issuing a publication.
3. Survey via the ISOE network conducted by Mr. Philippe Colson and Mr. Charles Pauron of EDF/DPN/CAPE/GPR.

# EVOLUTION AND CURRENT STATUS OF PERSONAL DOSIMETRY IN THE SLOVAK NPPS

**S. Mocko**
NPP Bohunice, Slovak Republic

**D. Viktory**
Public Health Office of the Slovak Republic

## History

In the archives of the Public Health Office, there are a lot of documents which demonstrate that since the beginning of design of the first NPP in Slovakia, there has been considerable attention dedicated to personal dosimetry. But only few facts will be mentioned here. The external exposure was monitored by national (Czechoslovak) dosimetry service in Prague until 1977. NPPs have had their own services since then. Film detectors have been used for legal purposes since the beginning. The structure of the film detector allows measurements of beta and gamma radiation, assessment of main direction and energy of radiation, identification of surface contamination. Exposed films are kept in the archive. The operational dosimetry has been developed significantly, from simple pen detectors with ionisation chamber to electronic dosimeters. The internal contamination has been monitored by medical service (under Health Ministry) which was established in NPP Bohunice until 1987. Since then the internal dosimetry services have been operated by the NPPs. The whole body counters and laboratories for analysis of biological samples have been available on both nuclear sites of Slovakia. The measurement system, the rules and the methodology were regularly improved. Quick body monitors and monitors for thyroid monitoring are available also on both sites at present.

## Current legal basis

The act on the health protection and the regulation of the Health Ministry on radiation protection determine the requirements for external and internal personal dosimetry of occupationally exposed workers.

Personal dose of any occupationally exposed worker of A category in controlled area must be monitored by personal dosimeter. All relevant components of radiation field on a workplace must be monitored by the basic (legal) personal dosimeter or dosimeters. Monitoring period for NPPs is one calendar month, but for less risk practices monitoring period up to three months is acceptable.

The extremity dosimeter is necessary if the extremity dose (or dose in lens of eye) could be significantly higher then personal dose monitored on the standard place for monitoring of personal dose equivalent. Direct readable operative personal dosimeter is obligatory in areas where the dose rates are higher than 1 $mSv.h^{-1}$ or where the radiation field changes rapidly so that accidental exposure is possible.

Requirements for the parameters and testing of personal dosimeters are based on technical recommendation of EC and the Safety Guide No. RS-G-1.1 on Assessment of occupational exposure due to external exposure.

Evaluation of exposure caused by internal contamination is obligatory on workplaces with radioactive substances. If the risk of internal contamination is not relevant, only area monitoring is obligatory. When the individual monitoring of internal exposure is required, the monitoring frequency and monitoring system depend on the risk of internal exposure.

The basic (legal) system for personal monitoring must be metrologically approved by the National Metrological Institute (stated by the act on metrology). Monitoring of personal doses in controlled area is obligatory. Only approved services can perform the basic (legal) personal dosimetry. The Public Health Office is responsible for approval and licence issuing for these services. In Slovakia there are three approved personal monitoring services for external exposure and two services for monitoring and evaluation of internal exposure at present.

### *Central register*

The central register of the occupationally exposed in Slovakia is established at Public Health Office of SR. This register co-operate with the register of radiation sources and licensees.

### Radiation passport

The radiation passports are not used in Slovakia at present. This is a weak point of the system of radiation protection and of the Slovak legislation. Slovakia will become a member of European Union and it is necessary to apply the directive on radiation protection of outside workers of EC. In the prepared amendment of the act on health protection, there should be the legal basis for issuing of radiation passports. I expect that national register will issue the passports.

### *Outside workers*

The operator is committed to ensure the same level of protection for the employees and for outside workers including the individual monitoring of external and internal exposure in controlled area. The operator is obliged to require the data about foregoing doses of the outside workers.

## Current status of the personal dosimetry in Slovak NPPs

### *Monitoring of occupationally exposed workers to external radiation*

There are two approved personal dosimetry services, one on site of Bohunice and the other one on site of Mochovce, both are operated by the Slovak Electric joint stock company. The personal dosimetry services were approved and licensed by the Health Ministry (in the future by the Public Health Office of SR). Film dosimeters are evaluated monthly (calendar month). The system of film personal dosimetry is metrologically approved by the National Metrological Institute in two years interval. The system of personal monitoring on site Bohunice and Mochovce differs, because of historical development. Any person in controlled area in NPP has a legal film dosimeter for gamma and beta radiation, electronic personal dosimeter for gamma radiation (or beta-gamma), and if necessary also neutron dosimeter and extremity dosimeter.

### *Basic (legal) personal dosimetry*

In NPP Bohunice, they use FOMA Personal monitoring Film (Czech Republic) (high and low sensitive). The cassette contains three Cu filters (thickness: 0.05, 0.5, 1.5 mm), Pb filter (0.5 mm) and a window (diameter 10 mm). Termoluminiscent dosimeters type LiF 600 are used for basic neutron dosimetry. Only very limited number of these dosimeters is necessary. Extremity dosimeters LiF 100 as finger dosimeters are used, if the extremity dose could be significantly higher.

In NPP Mochovce, the film dosimeter for gamma radiation contains Cu filters (0.05, 0.8 mm), Pb filter (0.2 mm) + Sn (0.6 mm), Cd filter (0.8 mm) and window without filter. They use also FOMA personal monitoring film R10 and R2, with high and low sensitivity. Neutron radiation doses are monitored by the TLD600/TLD700 in Bicron cassette. Extremity dosimeters – Aluminophosphate glass as finger dosimeters are used in special cassette.

Both personal dosimetry services use films irradiated in the National Metrological Institute (up to 30 films of the same rank as the film used in dosimeters irradiated with various doses) for the monthly calibration of densitometers (Gretag D 200 – II). The dosimeters are calibrated in $K_{a.}$ The conversion factors recommended by the ICRP 74 are used for the calculations. The quantities of $H_p(10)$ and $H_p(0.07)$ have been implemented since the beginning of 1993.

### *Operational personal dosimetry*

Any electronic personal dosimeter is calibrated in metrological laboratory of NPP annually. The metrological laboratory is approved by the National Metrological Institute and connected to the national standard.

NPP Bohunice: electronic dosimeters MGP, (DMC 90 and readers LDM 91) are used routinely for operational personal dosimetry (remote control WRM 91 is also available). The TLD system used for personal dosimetry is metrologically approved by the National Metrological Institute. The LiF 100 dosimeters are used, the Harshaw Model 6600 reader and card holders 8805.

NPP Bohunice is in the process of improvement of operational electronic personal dosimetry and replacement DMC 90 with the dosimeters of DMC 2000 type with new readers and calibrator.

NPP Mochovce: electronic dosimeters Siemens MARK 1 and MARK 2 (beta gamma) are used for operational personal dosimetry in controlled area and limited number of electronic neutron dosimeters is available. TLD system for operational personal monitoring consists of an aluminophosphate glass (diameter 8 mm, in plastic DIPRA cassette with perforated Pb filter) and reader SOLARO 680. The TLD system is regularly approved by the National Metrological Institute in two years interval.

NPP Mochovce intend to use the EPD as the basic monitoring system in the future. The authority requires to use dosimeters resistant to magnetic field and being able to monitor the week penetrating radiation only. It is also required to use simultaneously film and EPD systems for few years, an analysis how the transition to EPD will influence the system of radiation protection, mainly in case of higher exposure, because available EPD gives less information on energy and direction of radiation and surface contamination of dosimeter. It is also necessary to evaluate possibilities and conditions of safe data recording and storing.

### *Internal exposure monitoring*

Whole body counting, quick body monitoring for screening, $^{131}I$ in thyroid monitoring system and laboratories for analysis of biological samples are available on both NPP sites. Methodology for analysis of gamma nuclides in biological samples, tritium and strontium in urine and plutonium in faeces are approved.

Admission monitoring for any person who will work in controlled area (outside workers and new employees) is obligatory. Monitoring after retiring from work in controlled area (ending of the job on contract for outside workers, retiring of employees or retiring from work in controlled area) is obligatory. Routine monitoring – usually monthly obligatory for groups of workers working in enhanced risk of internal contamination. Periodical – annual monitoring is obligatory for all occupationally exposed (part of medical examination). Special monitoring – in case of anomalies, when the internal contamination is possible or expected.

Usually the highest relevant inhalation dose coefficient given in the BSS is used, if the chemical form of contaminant is unknown. Intake is calculated on the base of respiratory tract model (ICRP 66) and biocinetic models in ICRP 30.

### *Dose data management and record keeping*

Computer network system for work planning and dose management is available. It manages the data from basic dosimetry (film) and operational personal doses in the current month. After the evaluation of basic (film) dosimeter are the data on personal dose from operational dosimetry during last month are removed and superseded by the reading from basic dosimetry. The data from operational dosimetry are also kept in the archive.

All data on personal doses are archived on two different places in two forms – written and electronic. All measured values are recorded including those from operational dosimetry including data on methodology and important parameters. The data are regularly reported to the Central Registry and to the RP authority.

### *International comparison*

The personal dosimetry service of NPP Bohunice participated successfully in the international comparison for individual monitoring of external exposure from photon radiation organised by IAEA in 1997-1998. The personal dosimetry service of NPP Mochovce participated in international comparison of accidental dosimeters organised by Silene-Valdue. The results have not been published yet. Both personal dosimetry services took part in the international comparisons for whole body measurements and $^{131}I$ in thyroid measurements organised by IAEA. The results of intercomparison have not been published too.

### *Reporting system*

The dosimetric services and the licensee (for practice) are obliged to send the data on results of basic (legal) personal dosimetry during each monitoring period to the RP authority and to the central registry. They are also committed to provide analysis of personal dose (monthly and annually), annual and 5-years summaries. In the licence there is also specified what levels of individual doses (including those from operational dosimetry and internal dosimetry) should be reported without any delay to the authority.

## Conclusion

The good system of personal monitoring is the basic condition for appropriate radiation protection of occupationally exposed workers. Current system of individual personal monitoring of occupationally exposed workers in Slovak NPPs seems to be adequate and compatible with good practice. The main weakness of the system seems to be the missing system of the radiation passbooks. But we can expect that this problem will be solved till the end of this year.

**RADIATION RISK ANALYSIS OF TRITIUM IN PWR NUCLEAR POWER PLANT**

**Y. Maochun**
Daya Bay Nuclear Power Plant, Guangdong, PRC

**Abstract**

Tritium is a common radionuclide in PWR plant existing mostly in the manner of HTO, its radiation risk is mainly internal exposure when intake from inhalation. In this paper, the relationship between saturated HTO concentration in air (SHCA), HTO concentration in water ($A_{TW}$), and the water temperature was derived. Which is:

$$SHCA = 4.86 \times 10^{-6} \times 10^{\frac{6.9t}{230+t}} \times \mathrm{A_{TW}} .$$

In the normal operation of nuclear power plant, the practical HTO concentration in air (PHCA) is 30 to 60 times lower than SHCA. The radiation risk analysis of HTO revealed that, in PWR plant, the radiation risk of HTO is quite limited, no routine individual monitoring, no routine area or air monitoring and no special protection is needed for HTO.

Key words: radiation/risk/analysis/HTO/tritium.

**Radiation risk of tritium**

Tritium is a common radionuclide in PWR plant existing mainly in the manner of HTO, its radiation risk is mainly internal exposure when ingested by the following ways:

a) Absorption from skin when contaminated by Tritium. In a PWR plant, as effective individual protective measures are taken for contamination risk, large surface and high level skin contamination are normally averted. The possibility of Tritium intake by this way is quite low.

b) Intake from month. Because of the individual and collective protective measures implemented in PWR plant, the possibility by this way is low too.

c) Intake from inhalation: It is the main way of HTO ingestion.

HTO in water enter air mainly by evaporation. In the PWR plant, most of the radioactive systems are maintained enclosed in normal operation, except:

a) The spent fuel pool (in the fuel building): Always open to the air with surface of around 106 $m^3$.

b) Reactor pit and fuel transfer pool (in the reactor building): Filled with water in some periods of outage with a surface of around 150 $m^3$.

c) Liquid waste sumps: they are normally located in isolated rooms with small surface.

d) The reactor building during power operation: During power operation, the reactor building is maintained closed with only internal ventilation. As there is always some leakage collected in liquid waste sumps, after a certain time of operation, the HTO in air way be saturated with liquid and reach $10^4$~$10^5$Bq/m$^3$. Because access to reactor building is strictly controlled in power operation period, the internal exposure for workers from HTO is low.

In summary, the radiation risk of Tritium in RWR plant mainly exists in the fuel building, and the reactor building during outage. The highest HTO concentration in air is the saturated HTO concentration in air (SHCA) when equilibrium has been set up with water phase.

### Relationship between SHCA and HTO concentration in water ($A_{TW}$)

When the water temperature is low, there is:

1. $p_S \times V = n_S RT$

Here Ps is the partial pressure of saturated steam in air, $n_s$ is the mole number of saturated steam in air, R is a constant, R=8.31 J/mole.k, V is the volume of air, T is the absolute temperature.

For HTO in air:

2. $p_{TS} \times V = n_{TS} RT, \ \frac{n_{TS}}{V} = \frac{p_{TS}}{RT}$

Here $P_{TS}$ is the partial pressure of HTO in the saturated steam, $n_{TS}$ is the mole of HTO in air.

The activity of HTO in air:

3. $A_{TS} = \lambda \times N_{TS} = \lambda \times \eta \times n_{TS}$

Here, $n_{TS}$ is the number of HTO molecules, $\lambda$is the decay constant of HTO, $\eta$is a constant, $\eta=6.023\times10^{23}$

The saturated HTO concentration in air (SHCA) when equilibrium is set up between air and water is:

4. $SHCA = \frac{A_{TS}}{V} = \frac{\lambda \times \eta \times n_{TS}}{V} = \frac{\lambda \times \eta \times p_{TS}}{RT}$

If the HTO concentration in water is $A_{TW}$(Bq/m$^3$), the mole of HTO is $n_{TW}$, there is:

5. $A_{TW} = \lambda \times n_{TW} \times \eta$

6. $n_{TW} = \frac{A_{TW}}{\lambda \times \eta}$

7. $n_w = \frac{10^6}{18} mol = 5.56 \times 10^4 mol$

In 1 $m^3$ water, the number of mole of $H_2O$ is:

$$8. \quad \frac{n_{TW}}{n_W} = \frac{A_{TW}}{\lambda \times \eta \times 5.56 \times 10^4} = \frac{1.8 \times 10^{-5} A_{TW}}{\lambda \times \eta}$$

So the relationship between $P_{TS}$ and Ps is:

$$9. \quad p_{TS} = \frac{n_{TW}}{n_W} \times p_S = \frac{1.8 \times 10^{-5} A_{TW} \times p_S}{\lambda \times \eta}$$

When (8) is combined with (4), there is:

$$10. \quad SHCA = \frac{\lambda \times \eta \times p_{TS}}{RT} = \frac{1.8 \times 10^{-5} A_{TW} \times p_S}{RT} = \frac{2.17 \times 10^{-6} A_{TW} \times p_S}{T}$$

The partial pressure of saturated steam in air (Ps) has the following relationship with water temperature:

$$11. \quad \mathrm{Ps} = 2.24 \times 10^{\frac{6.9t}{230+t}} \times T$$

As a result:

$$12. \quad SHCA = 4.86 \times 10^{-6} \times 10^{\frac{6.9t}{230+t}} \times \mathrm{A}_{TW}$$

According to (12), at the typical temperatures, the relationship of HTO concentration in air (SHCA) and in water ($A_{TW}$) are:

13. $SHCA_{20\,°C} = 1.73 \times 10^{-5} A_{TW}$

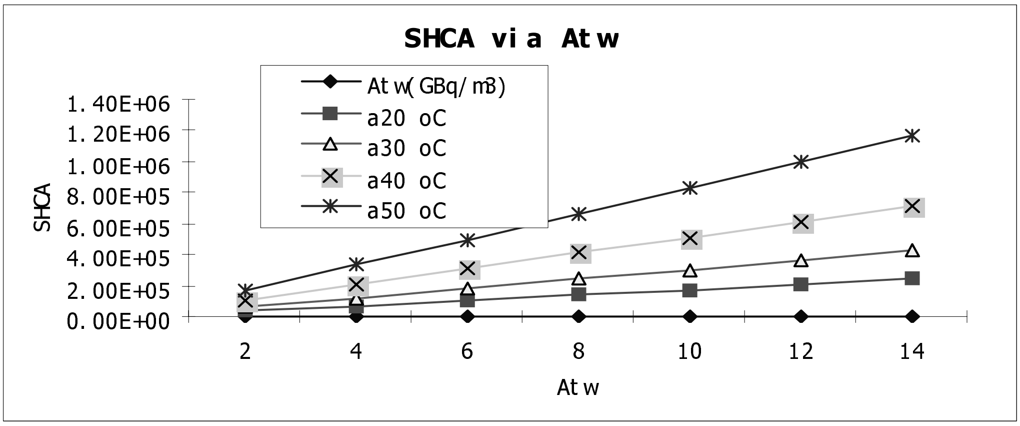
14. $SHCA_{30\,°C} = 3.04 \times 10^{-5} A_{TW}$

15. $SHCA_{40\,°C} = 5.12 \times 10^{-5} A_{TW}$

16. $SHCA_{50\,°C} = 8.30 \times 10^{-5} A_{TW}$

The above relationship could be illustrated in Figure1.

**Figure 1. SHCA via Atw**



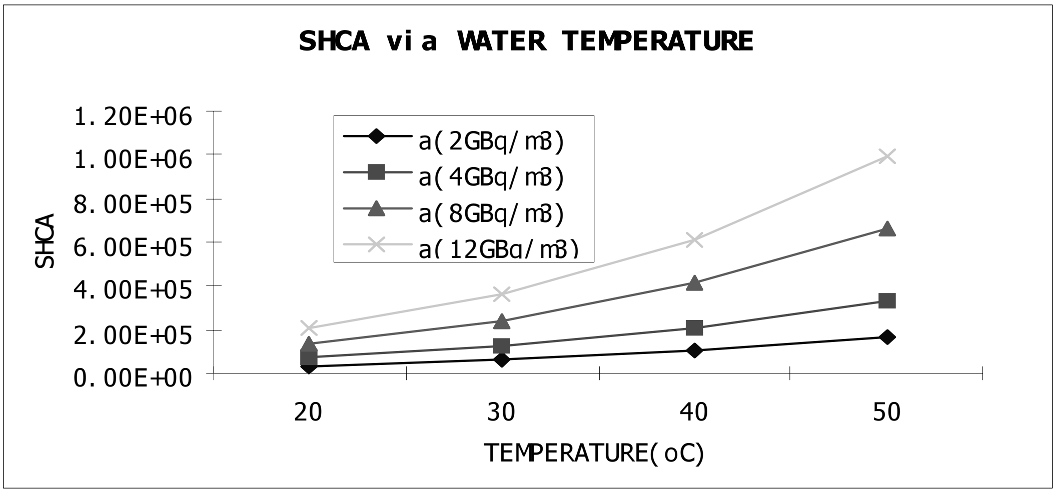
For a PWR plant, when the purification and residual heat removal system for the reactor pit and spent fuel pool is in normal operation, the designed highest water temperature is normally 50°C, the practical temperature is normally 30°C. For respiratory intake, the DAC of HTO is $8\times10^5$ Bq/m$^3$. According to Figure 1, when the water temperature is 30°C and the HTO concentration ($A_{TW}$) is 3GBq/m$^3$, the *SHCA* reaches 0.1DAC, when $A_{TW}$ is 8GBq/m$^3$, *SHCA* reaches 0.3DAC, when $A_{TW}$ is as high as 28GBq, *SHCA* reaches 1DAC.

According to equation (12), at different $A_{TW}$, for example, when $A_{TW}$=4GBq/m$^3$, the relationship between *SHCA* and water temperature is:

17. $$SHCA = 1.94\times10^4 \times 10^{\frac{6.9t}{230+t}}$$

Figure 2 is the relationship between *SHCA* and water temperature at different level of $A_{TW}$.

**Figure 2. Relationship between *SHCA* and water temperature at different level of $A_{TW}$**

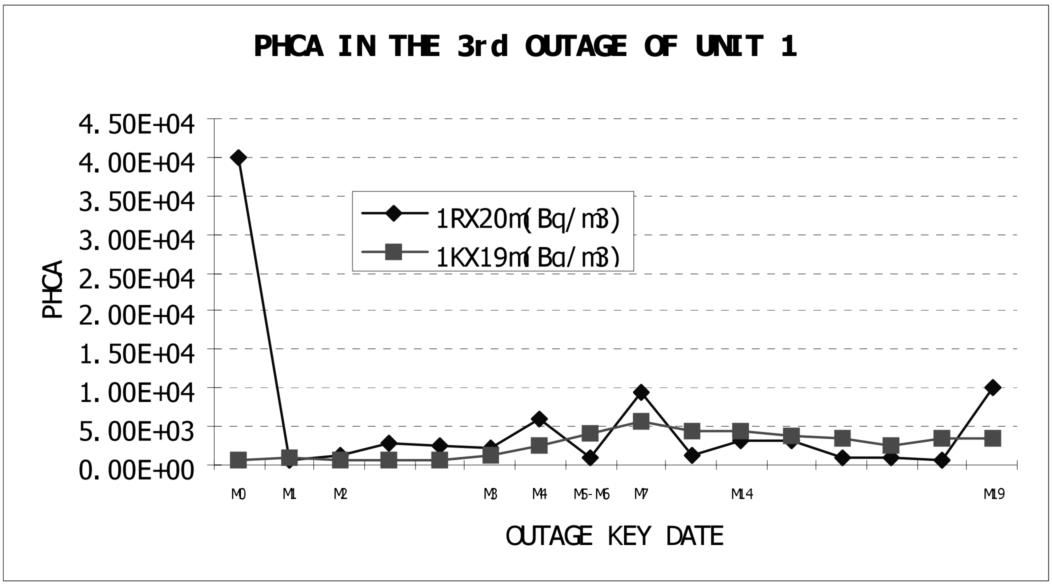



From Figure 2, the equilibrium HTO concentration in air increase evidently with water temperature, when water temperature is 50°C, the *SHCA* is 2.7 times to that of 30°C, and 4.8 times to that of 20°C. when $A_{TW}$= 8GBq/m$^3$, t = 20°C, *SHCA* = $1.4\times10^5$Bq/m$^3$, t = 30°C, *SHCA* = $2.43\times10^5$Bq/m$^3$, when temperature reach 50°C, *SHCA* shall reach 1DAC.

**The practical radiation risk analysis of HTO**

Figure 3 is the practical HTO concentration in air (PHCA) over the reactor pit in the reactor building (1RX) and over the spent fuel pit in the fuel building (1KX) in the 3rd outage of unit 1 of the Daya Bay Nuclear Power Plant.

**Figure 3. PHCA in the 3rd Outage of Unit 1**




The X axial is the key date of the outage. Mo is the date of unit shutdown for outage, M1 is the date that unit reach cold shutdown, M2 is the date when the pressuriser manhole was opened, M3 is the date of unloading, M4 is the date the fuel transfer ended. Between M5 and M6, the reactor pit is empty to facilitate maintenance on the primary circuit. M7 is a specific period in the outage for reactor building pressure test, $M_{14}$ is the refuelling period, $M_{19}$ is the end of the outage and the unit reached hot shutdown status.

According to the practical monitoring results of $A_{TW}$ in the outage, the $A_{TW}$ of spent fuel pool was around 3GBq/m$^3$, while the reactor pit was around 6GBq/m$^3$. Suppose the water temperature was maintained at 30°C, the SHCA in 1KX and 1RX shall be $9.12\times10^4$Bq/m$^3$ and $1.82\times10^5$Bq/m$^3$, the PHCA was around 3KBq/m$^3$, which is 3.23% and 1.65% of the SHCA.

Another factor affecting the HTO concentration in air is the $A_{TW}$. The practical results of DaYaBay NPP in the past 8 years demonstrate that the HTO concentration in primary coolant was about 50GBq/m$^3$ during power operation, and less than 20GBq/m$^3$ in the outage. As a result, the HTO in the spent fuel pool and the refuelling water tank shall normally not be more than 20GBq/m$^3$. If HTO concentration in air is 50 times lower than its SHCA when ventilation system is in operation, at 30°C, the PHCA shall not reach 1DAC except when the $A_{TW}$ reaches 1.3TBq/m$^3$.

## Conclusion and proposals to HTO monitoring and protection

1. There are only limited areas existing radiation risk in RWR Nuclear Power Plant, with the practical situation that ventilation in operation, low water temperature and lower $A_{TW}$, the radiation risk of HTO is quite low.
2. For individual protection, no special protection is needed for HTO in PWR plants.
3. For individual dose monitoring, except the monitoring for selected samples of workers, no routine monitoring is needed.
4. For the area monitoring, except the special monitoring, no routine monitoring is needed.

5. Maintain normal operation of the ventilation system and the spent fuel pool cooling system is needed and effective to limit HTO in air.

**Reference**


1. *Radiation protection to Tritium, Yang Huaiyuan*, Nuclear energy publication, Beijing, PRC, 1996.
2. *The HTO monitoring and risk analysis in Daya Bay* Nuclear Power Plant (report not published), Yang huaiyuan, 1997.

# OPERATIONAL EXPERIENCE WITH A LEGAL ELECTRONIC DOSIMETRY SYSTEM

**T. Zodiates**
British Energy, United Kingdom

## Abstract

The Thermo (Siemens) Electronic Dosemeter EPD-1.2D is the Approved dosimetry system for British Energy (BE). It has been in use for a number of years firstly as a control dosemeter and latterly as a legal dosemeter. This paper reviews its performance from a corporate perspective and considers issues such as maintenance strategy, sharing of dosemeters across a number of sites and common user problems encountered across sites.

## Introduction

British Energy (BE) and BNFL/Magnox are using the Thermo Electronic Personal Dosemeters (EPD) as the legal dosemeter in a number of their stations. Initially the EPD was used as a control dosemeter. BE decided to embark in the transition from legal passive dosimetry to the legal EPD in 1998.

Approval for use of the EPD (Mk 1) as a legal dosemeter was obtained in 2000. The first UK station to use the EPD as the legal dosemeter was the Magnox Station Oldbury in 2000. The first BE station was the Dungeness B AGR in February 2001. Currently six out of the eight BE power stations use the EPD as the legal dosemeter.

At present BE's Approved Dosimetry Service operated by BNFL has applied to the Health and Safety Executive (HSE) for approval of the EPD Mk 2 for use as a legal dosemeter. BE's Hinkley Point B will be the first client of that service.

The purpose of this paper is to review the operational experience with the EPD from a corporate perspective.

## The EPD system

The EPD system implemented in the BE stations is based on the EPD hardware and software developed by the then Siemens Environmental Systems Ltd, currently part of the Thermo Electron Corporation (Thermo). The main components of the system and their purpose are:

- EPD: this is the dose measuring device (dosemeter) and is worn by the workers;
- EPD system: this is a computer based network consisting of the server, the access control terminals (ACT), and the user workstations;
- software "dose control system" (DCS).

The EPD system also interfaces with the approved dosimetry service (ADS) and the station security system for access into the radiological controlled area (RCA).

**Transition from passive to electronic dosimetry**

At the beginning of the process the stations used the film badge as the legal dosemeter and various electronic dosemeters as control dosemeters. These electronic dosemeters were in general issued and read manually.

The move to using the EPD, with its required hardware and software infrastructure, was a significant change to the workers customs and practices in place at the time. It also required a significant increase in the support to the dosimetry systems from the station information technology (IT) departments. The increase in the complexity of the dosimetry system, which was to become effectively a computer system, also raised concerns with regards to its management and integrity.

A working group with representatives from all the company stations was set up to co-ordinate the transition phase. It was agreed that each station would proceed at its own pace to ensure that the system gained acceptance by the station workers and by the other departments. The working group would also agree standard approaches to be implemented at all stations as far as practicable.

The transition to the Thermo EPD as the legal dosemeter took place in three phased steps:

1. introduce the EPD as a control dosemeter on restricted issue; i.e. issued as required by the work permit;
2. use the EPD as a control dosemeter on general issue; i.e. issued to all that enter the RCA;
3. use the EPD as the legal dosemeter.

To assist each station through this transition, a standard quality plan was developed covering issues like:

- procurement, installation, interfacing and testing of the EPD system;
- development of EPD system documentation: work instructions, procedures, disaster recovery plans;
- training of EPD users and dosimetry office staff;
- liaison with local Health & Safety committees and regulators;
- system audits by the ADS and HSE.

The transition phase progressed without any major problems. The main difficulties encountered were:

- gaining confidence with the EPD dosemeter due to its teething problems in particular battery problems and the ongoing RFI interference with the Mk 1;
- ensuring that the workers look after the EPD; initially there were too many EPDs damaged by being knocked off surfaces or dropped from pockets/belts;
- obtaining the necessary computing support from within the limited station resources; for example not all stations have Oracle expertise.

## Organisational arrangements

The EPD system is a dosimetry system and as such comes under the control of the station Health Physics department. The driving force of the system at each station is the Dosimetry Officer and the day to day operator of the system is the dosimetry administrator.

## Hardware management

The hardware management covers the dosemeters and the EPD/computing hardware (ACT, servers, etc.).

### *Dosemeters*

Each station is equipped with sufficient EPDs to support the day to day normal operation requirements. The stations operate a pool of additional EPDs to supplement their requirement during outages when there is a need for more EPDs.

The logistics of the EPD management to ensure that there is the right number of EPDs at each station whilst keeping the total number of EPDs low is very challenging. This has to consider not only the EPDs that are available for use, but the potential number of EPDs not available either because they are out for repair or have been sent for calibration. The distribution of EPDs in summer of 2002 with 2 outages taking place was:

| **EPD Mk 1 distribution in Summer 2002** | | | | |
|---|---|---|---|---|
| **Station** | **In use** | **Emergency packs** | **Off site** | **Pool** |
| **SXB** | 132 | 40 | | 150 |
| **DNB** | 435 (outage) | | | |
| **HYA** | 150 | 70 | 24 | 90 |
| **HYB** | 150 | 67 | | |
| **HNB** | 300 | | 13 | |
| **HRA** | 400 (outage) | 50 | 41 | |
| **Total** | **1 567** | **227** | **78** | **240** |

The total number of EPDs to support the six stations (5 AGR and 1 PWR) is about 2 100.

The typical numbers of EPDs used at the stations are:

- Normal operation: 150
- Outages: AGR: 350-400 PWR: 450
- In emergency packs: 50-80

The EPD Mk 1 batteries cost about £15-£16 (£ English Sterling) each. In order to limit the costs a lead station buys batteries in bulk at better rates and each station replenishes their stock from the lead station as and when required.

The ability to do basic EPD repairs varies from station to station. Typical repairs that a station may carry out are replacement of batteries, buttons and clips. Recently replacement of seals/RFI gaskets has been carried out at Heysham to improve the RFI performance of the EPD.

***EPD related hardware***

Each station has its own compliment of EPD hardware. At present there is no central holding of spares although it is being considered as part of the future deployment of the Mk 2. Each station has sufficient ACTs, desktop readers and access control units (interface with security system) so that unavailability of one or two individual units is operationally acceptable. Typical numbers of equipment are:

- Access control terminals (ACT): 8-12;
- Desktop readers: 2;
- Access control units (ACU): 4-6.

The main critical component of the EPD system is the server, which operates the whole system. The initial approach considered was to have two spare servers at a central location that can be dispatched to where they are needed. Due to the different arrangements and types of servers at each station, in the end it was decided to have a spare server at each station.

As part of the approval for the EPD system there is a need to have detailed disaster recovery plans for the system. These cover the arrangements from a malfunction of a single issued EPD, to failure of the system server or network.

One of the major issues that were addressed is the possibility of server failure. Such a failure can result in the loss of the database. Therefore there is a data back up procedure in place that ensures the loss of data is minimised. Another important aspect of the system is the ability of the ACT to operate in a standalone mode for up to 48 hours. The ACT set up is such that, if the server is lost or communication fails, the ACTs continue to operate in local mode issuing and returning EPDs. This enables the work to continue whilst the system is being restored. Following restoration of the system, the data stored on the ACTs is uploaded to the server. There have been three instances of server failure and in all three the EPD system operated successfully.

***System support arrangements***

Until recently all support to the system was provided through a hardware maintenance and software support contract with the EPD supplier (Thermo). Recently (2004), first line IT system support is provided from the Company's IT department (see later).

Failed hardware is sent to Thermo for repairs and is then returned to the station. For software issues, Thermo operates a Helpdesk for reporting software problems and queries. Response to the calls depends on the severity of the problem.

These arrangements have been successful over the years and the service provided has been very good. In particular, the Thermo response in the few real station EPD emergencies was excellent.

EPD dosemeters are calibrated annually by a Calibration Laboratory with accreditation to ISO 17025.

***Operational experience***

The introduction of the electronic dosemeter in such a large scale resulted in a number of issues being raised covering the end user, the dosemeter and the whole system.

**End user experience**

The end user issues relate to the use of the EPD by the workers and health physics staff. They relate to the practices that are used in each station. Some of these became issues because of itinerant workers experiencing different practices at different stations. Typical issues are:

- Training: a generic training package was developed for use that had sections to incorporate local station practices. This resulted in itinerant workers receiving the training many times and having to adhere to the local way of using the EPD. Over the period we have continued the progress towards standardisation to reduce these station differences.
- Method of wearing the EPD in relation to different types of overalls or personal protective equipment; this was important in order to reduce damage to the dosemeters. A "pouch" was introduced to stop EPD dropping off from pockets.
- Changes in dosimetry work: Dosimetry work changed from being a "hands on" job managing film badges to managing a software system. This required a change in the knowledge and expertise of the dosimetry staff. There was a certain degree of reluctance and trepidation towards the change. Networking between the station dosimetry staff helped the exchange of information and helped with any problems arising.
- User confidence: The EPD Mk 1 had a number of "teething" problems, mainly battery and radio frequency interferance (RFI) problems. The EPD Mk 1 is still susceptible to RFI. These have delayed acceptance of the EPD by the users.

Overall, there have been no significant issues associated with the end user except of those arising whenever a change is implemented. The gradual transition process and the parallel use of EPD and film badge helped in the acceptance of the EPD by the end users. The acceptance of the EPD was likened to the introduction of wearing seatbelts in a car; it has now been accepted as the "norm".

**EPD dosemeter experience**

A number of different problems arose with the EPD dosemeter over the years. Some of them can be attributed to the user such as mishandling (vandalised, dropped, knocked) or not following instructions during issue/return of EPDs. Other faults are typical faults expected from such equipment such as problems with the audible alarm, the display, or the detectour.

Numerous failures of EPDs resulting in loss of visit dose data have occurred due to the mishandling of the EPD, battery failure, RFI etc. In such an event, a dose assessment is carried out by the Health Physicist, and a record is entered in the database. This has resulted in additional Health Physics effort and resource being required to support the system.

Overall, there were three significant EPD Mk 1 problems. These were:

- Battery problems: early battery failures or battery/EPD circuit board connection failures; this has been addressed by improving the connections and the battery supply chain.

- RF interference (RFI): this is a persistent problem inherent in the Mk 1; experience with the EPD Mk 2 at Hinkley Point B has shown that in 400 000 individual visits since August 2000 there has been only one spurious dose assessment (due to proximity to a welding set).
- Beta window failure: this failure is due to degradation of the seal between the EPD case and the beta window resulting in in-leakage of light which results in a false reading of high beta dose. Although the root cause of the problem has not been eliminated, the cause and effects are now known and they can at least be managed.

The RFI interference is generic to the Mk 1 EPD design and is being managed procedurally at the stations. It still provides a significant workload to the Health Physicists in terms of dose investigations. The introduction of the EPD Mk 2 will eliminate this problem.

Two generic problems were experienced with the EPD Mk 2:

- Cracking of the lids: a manufacturing defect caused the EPD lids to crack at the corners; the manufacturing process was corrected and all the lids were replaced by the supplier.
- Flexi connector (power supply circuit board connection) failure: environmental factors (temperature variations) could cause failure of the flexi connector and hence loss of circuit board power supply. The power supply connections were re-engineered and all the EPDs were repaired by the supplier.

**EPD system (hardware & software) experience**

Although a number of hardware failures and software problems occurred, overall the EPD system has been remarkably robust both in terms of hardware and software. There are two issues worth highlighting.

***Resource required for computing support***

At the beginning of the project there was a failure in ensuring computing support to the station Health Physics from the Company’s central IT departments. The reasons for that were many and varied. The consequence was that stations did not have sufficient expertise to support the EPD project. In addition the station computing infrastructure was different from station to station and this lead to the EPD system being implemented in different types of networks such as stand alone networks, virtual networks, and networks integrated with other station networks. This lead to a number of different problems experienced at different stations, such as network communication problems, data back up differences, and in user support.

In the case of Hinkley Point B, the installation of the EPD system was driven by a project team which included IT specialists from the central IT department. The system as installed has been exceptionally stable. It has also incorporated a “Citrix” interface for providing access for numerous users without the need to install the necessary software on the user’s own PC.

Over the last 18 months these problems have been overcome by putting in place a project to provide central computing support to all the stations. This includes standardising servers, data back-up arrangements, network configuration, Windows and Oracle support, security and others.

***EPD issue/return faults affecting use's perception***

There have been a handful of occasions where due to errors in the EPD issue or return process the visit data was not properly recorded and the EPD remained issued to the user without the user realising. These resulted in visit data being lost and problems with subsequent user attempts to self-issue another EPD.

In terms of dosimetry, the impact of these faults was small requiring a dose assessment to be made for the specific worker visit. Compared with the loss or damage of a film badge the loss of data of a single EPD visit is insignificant.

However, these few events resulted in widespread concern about the integrity of the system as rumors spread between the stations. The concerns and rumours died out slowly with the good performance of the system.

**System hardware and software development**

The EPD Mk 1 was developed in the early 1990s. Sizewell B deployed the Mk 1 as a control dosemeter from the beginning of its operation in 1994/95. The original system software, DCS, was developed to support the EPD application at Sizewell B.

**Hardware development**

The EPD Mk 1 hardware has remained virtually unchanged since its development. Thermo will be discontinuing the Mk 1 support in 2004 due to obsolescence of components forcing them to withdraw the maintenance support of the system. However, the EPD Mk 2 has been developed to replace the Mk 1. The Mk 2 incorporates significant improvements. From an end user's point of view the most significant are:

- it overcomes the RFI problem;
- it is smaller, lighter and more robust;
- it uses commercially available batteries (Mk 1 requires specialist bespoke battery);
- it can provide more detailed dosimetric information for the investigation and assessment of doses. This requires a suitable upgrade of the dose control software.

The Mk 1 to Mk 2 transition process has financial and logistical implications. The current plan is to phase in the Mk 2 over a 3-4 year period in order to spread the cost and maximise the use of the Mk 1.

One of the future developments that are being considered with the introduction of the Mk 2 is the move to a single server/database for all the stations. The advantages of this system are:

- the information for each worker needs to be entered once and it is then available to all stations;
- the up-to-date total worker dose is available in real time especially for itinerant workers;
- training and other compliance issues become easier to manage.

The disadvantages of such a system are: the robustness and speed of the computing network to support such a system without undue delays especially at the EPD issue/return point constraints on the EPD and DCS3 software to support the way the EPD and other data is used.

## Software development

Software developments have been required for a variety of reasons. There have been user driven developments and there have been imposed developments.

As the users became more familiar with the system and the system capabilities became obvious to the users, there has been a demand for improvements in the user-system interface and for additional services. The user developments took the software through the following main versions:

- DCS2; 1998/99: add necessary functionality for legal system and to incorporate other dosimetry systems for millennium compliance (e.g. replace the film badge management software);
- DCS2-SP1; 1999/2000: interface with the ADS measurement; fix minor bugs; additional user services. This was the first software version to be used for the legal EPD at Dungeness;
- DCS3; 2000/02: additional user services.

The latest software version is DCS3 V2.3 and is about to be deployed to the BE stations. In parallel with the BE/Magnox support, Thermo is developing DCS4 for another UK user. BE and Magnox have been kept involved in this development with the aim of making DCS4 the standard software used in the United Kingdom.

The biggest driver for software development, however, are the operating software suppliers (Windows and Oracle). Decisions on whether to upgrade the operating software or to continue using previous versions are difficult. Based on our experience with the EPD software and its robustness we continue using the Windows NT and Oracle Version 7 software. However, this is becoming untenable and a BE corporate decision has been taken to move to Windows XP and Oracle 9i over the next three years.

There is a need to upgrade the software at the same time as changing from the Mk 1 to the Mk 2. BE is currently considering how best to achieve these targets.

## Conclusions

The introduction of electronic dosemeters for legal use within BE has been successful. The Thermo EPD Mk 1 is now used routinely and has gained a high level of acceptance by the workforce. The main issues that require attention with electronic dosimetry are:

- the reliability and ruggedness of the dosemeter;
- the arrangements for the computing support of the system; and
- obsolescence and updates of components and software.

# SESSION IV

# SAN ONOFRE UNIT 1 DECOMMISSIONING

**E.M. Goldin**
Southern California Edison, United States

## Introduction

Nuclear plant decommissioning presents several challenges in radiation protection. The plant demolition must consider radiation protection for workers, protection of the public and careful material management. Decommissioning of the San Onofre Nuclear Generating Station (SONGS) Unit 1 presented some additional challenges.

### *SONGS 1 history*

SONGS 1 was a first generation Westinghouse 3-loop pressurised water reactor (PWR) originally rated at 450 MWe. The unit operated from January 1968 to November 1992 when it was permanently retired from service. Containment consisted of a 2.5 cm thick steel sphere. In 1976, a 1 m thick steel-reinforced concrete Sphere Enclosure Building was constructed around the sphere for post-accident radiation shielding. One of the more unique aspects of SONGS 1 is that it is collocated with SONGS Units 2 and 3 that are newer 1 100 MWe Combustion Engineering reactors. They were declared commercial in 1983 and 1984 respectively.

The unit was permanently retired after 25 years of service due to financial considerations. To bring the plant up to the more strict safety standards of the more modern nuclear plants, a number of plant modifications were still required. Rather than invest that capital in the plant, the regulators and company agreed to shut the unit down. At the time, it had 15 years left on the operating license.

Once shut down, the US Nuclear Regulatory Commission (USNRC) requires that plants either begin immediate dismantlement, known as DECON, or be placed in a condition known as SAFSTOR. In SAFSTOR, fuel is removed from the reactor and systems are retired that are no longer needed to maintain safe cooling of the irradiated fuel. SONGS 1 was placed in SAFSTOR in 1993. The operating license was converted to a possession-only license. Systems were categorised into those required for operation (RO) such as spent fuel cooling and those not required for operation (NRO). The intention was that the unit would remain in a SAFSTOR configuration until the permanent retirement of Units 2 and 3, projected for many years in the future.

In the late 1990s the decision was made to begin active dismantlement of the plant. Decommissioning will result in reduced customer costs through lower fuel storage costs. The spent fuel will be placed into dry cask storage, an independent spent fuel storage installation or ISFSI. At the time, low-level radioactive waste (LLRW) disposal costs were also known whereas LLRW disposal in the future was uncertain. Dismantlement could also be accomplished safely using proven technologies. Moreover, there are personnel still working at the site who are familiar with SONGS construction, design, and operation. And finally, there are sufficient funds available in the decommissioning trust.

***Project priorities***

The primary goals of the project included:

- protection of the spent fuel throughout the decommissioning process until the US Federal Government (Department of Energy) accepted the fuel for disposal;
- industrial health and safety;
- disposal of radioactive and hazardous wastes according to the highest standards practical to ensure long term public health and safety;
- compliance with all applicable state and federal requirements; recognition by the public of that compliance;
- perform the work in a reasonable and prudent manner.

***Major activities***

In addition to the complication of having two large operating units on the same site, the SONGS 1 decommissioning project was also constrained due to the very small site. There is little room for laydown space for staging equipment, waste containers, rubble, offices, etc. Very careful planning was required for that reason in addition to the normal demolition planning. Moreover, with little space, radiation levels around the work area varied depending on the location of materials being removed. These varying levels had to be considered when conducting radioactive contamination surveys of nearby materials.

One of the early projects was the separation of Unit 1 from the security area for Units 2 and 3. Since Unit 1 was built first, a number of systems had to be separated from the plant since they supported the operation of the two larger units. These included the meteorological tower, some electrical supplies, and fire protection.

The unit was provided with independent monitoring and isolated electrical and water supplies. This allowed the majority of the existing plant to be declared “cold and dark”. That meant that demolition crews could cut into piping and electrical systems without worrying if the systems were still in service. Active systems were identified with bright orange paint.

Some of the first buildings demolished included the emergency diesel generator building (the diesel generators had been removed and sold) and the control building (a new independent control room was constructed). The sequence of major building removal was developed to make room for the ISFSI.

License termination would normally take place after complete dismantlement of buildings and structures and restoration of the property. However, that will probably not occur until many years into the future simply due to the existence of the ISFSI and the adjacent operating units.

### *ALARA programme*

Decommissioning presents some unique challenges for an ALARA programme. However, the majority of the work may be accomplished using existing ALARA programme elements. Therefore at SONGS, the ALARA programme is considered "sitewide", applying to the decommissioning unit as well as the operating units. Those programme elements include:

- job planning;
- dose controls, administrative limits;
- application of temporary shielding if appropriate;
- pre-job briefings;
- dose estimates that serve to identify priorities, establish goals and monitor performance;
- use of mock-ups and training for specific high-dose jobs.

There are some specific issues that are important at a decommissioning plant. Foremost is the removal of the high dose components first. For SONGS 1 that included several components at the lowest elevation of the plant such as residual heat removal equipment and removal of the regenerative heat exchanger that presented a high source term in an open area in containment.

Temporary shielding was used when the dose saved exceeded the dose expended to install and remove the shielding. Special instructions were developed for Unit 1 because of the reduced requirements for installing heavy lead blankets on components that were no longer going to be needed for plant operations. Greater loading was available and a much simpler approval process was developed.

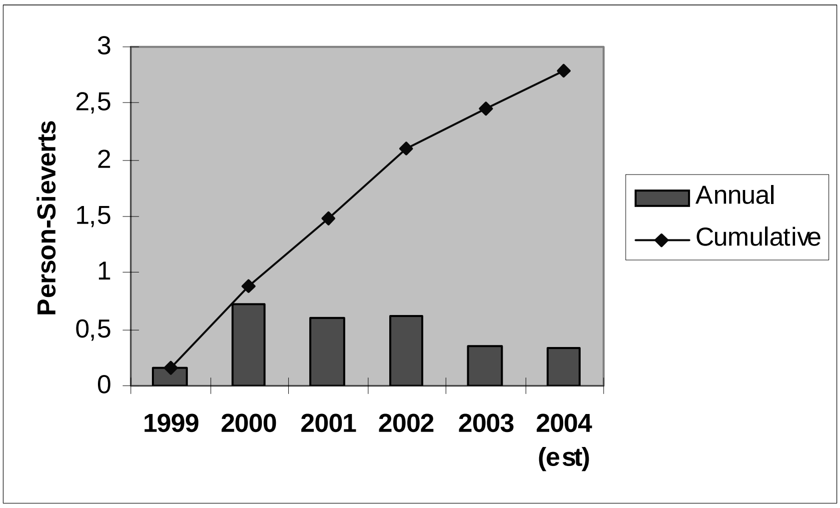
Airborne contamination becomes very different at plants that have been shutdown for at least several years. The decay of most of the shorter-lived beta/gamma emitting radionuclides leaves an increasing contribution from alpha-emitting radionuclides such as the transuranics. For an air sampling programme, the reduced contribution of beta/gamma emitters means that more care is required to distinguish between naturally occurring alpha-emitters and plant related contamination. We developed a protocol to facilitate prompt identification of airborne contamination with follow-up counts to distinguish natural radioactivity.

### *Major projects completed to date*

The table below presents cumulative radiation exposure in person-sieverts. "HP" indicates radiation protection activities that includes job coverage, surveys, waste packaging, etc.

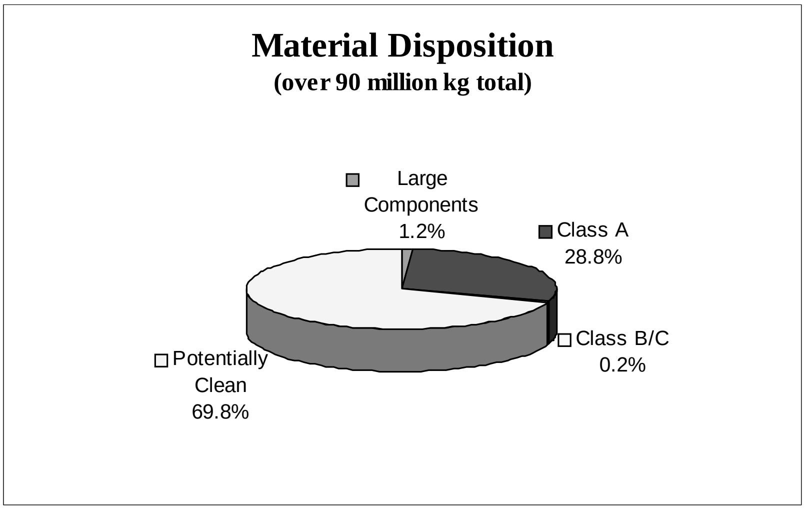
| YEAR | PROJECT | Person-Sv |
|---|---|---|
| **1999** | Regenerative heat exchanger | 0.061 |
| | HP functions | 0.068 |
| **2000** | Reactor coolant system severance | 0.158 |
| | Remove reactor head superstructure | 0.058 |
| | Remove interferences and system equip. | 0.069 |
| | Support (scaffolding, temp power) | 0.076 |
| | Health physics (HP) functions | 0.171 |
| | Asbestos abatement | 0.095 |
| **2001** | Large component removal preps | 0.136 |
| | Reactor vessel internals (RVI) segmentation | 0.179 |
| | Large component removal and RVI support work | 0.114 |
| | HP functions | 0.149 |
| **2002** | Large component removal | 0.358 |
| | RVI segmentation | 0.049 |
| | Support work | 0.080 |
| | HP functions | 0.115 |
| **2003** | Containment systems removal | 0.215 |
| | HP functions | 0.060 |
| | Fuel transfer | 0.033 |
| | Containment decontamination | 0.029 |

The cumulative radiation exposure for the entire decommissioning project will likely total about 4.5 person-Sv. The graph below indicates performance at about half project completion. Note that the annual exposure will continue to decrease as more and more of the sources are removed.



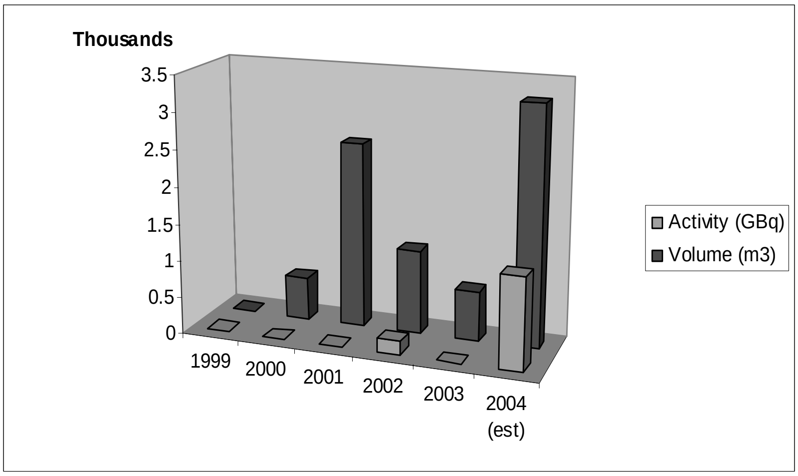
*Materials management*

One of the larger focuses of the project is the efficient disposal of waste materials. This includes the very careful distinction between what is radioactively contaminated and what is not. Early in the project planning phase, estimates were made for the total quantities of materials including estimates of the various low-level waste classifications and the relative amounts of clean materials. The pie-chart below depicts the estimated quantities.




*Waste disposal*

Radioactive waste shipments to date are shown in the following bar chart. Note that the dominant contribution to the total radioactivity is the reactor pressure vessel and internal components that were anticipated for disposal in 2004 (but will not be). Also in 2004, a large volume of contaminated rubble from inside containment is expected.



### *Large component removal*

From a radiological perspective, an early goal was to remove the large components – the reactor pressure vessel, the three steam generators, and the pressuriser. A first step in large component removal was the segmentation of the reactor internals. Due to limitations of the total radioactivity quantity acceptable by the disposal facility, those internals had to be sectioned so that some parts could be placed in the reactor vessel for ultimate disposal and the higher activity parts were removed for long-term storage in the ISFSI. This particular job had the potential for significant radiation exposure based on experiences at other plants. Therefore, considerable effort was expended to ensure best practices, use of reliable equipment, and mockup training when appropriate.

The large components were lifted out of the containment sphere after the SEB roof was removed and holes were cut into the sphere. Restrictions were placed on work inside containment during the large component lifts to ensure that contamination was not stirred up to present potential airborne releases out the openings. After the large components were removed, covers were placed over the openings to prevent rain intrusion and to minimise release paths for contamination.

The three steam generators and the pressuriser were shipped for disposal by rail. Due to size limitations, the reactor head was shipped using an oversized truck. The reactor pressure vessel was packaged with some of the internals within a steel canister. Low-density grout was placed for stability both inside the vessel and between the vessel and the canister. Other components that presented shipping challenges due simply to size and weight included the three reactor coolant pumps. Once all those components were removed along with a few smaller components inside containment, the radiation levels to workers were greatly reduced.

## Conclusions

Decommissioning including complete removal of above ground structures can be accomplished safely and efficiently. None of the low-level radioactive waste is unique to decommissioning although transportation of large components can be a significant challenge. Proven techniques are available to handle Greater than Class C waste (highly activated reactor internals) and spent fuel.

A considerable challenge is the dispositioning of the very large volume of potentially clean material. There is a high cost to survey and decontaminate materials. Moreover, in the United States today there are no standards for the clearance of potentially contaminated volumetric materials. Careful planning is necessary to determine the most cost-effective means for waste management, whether it includes decontamination and surveys or simple disposal.

Lastly, existing ALARA programmes with some minor modifications provide sufficient worker and public protection from radiation.

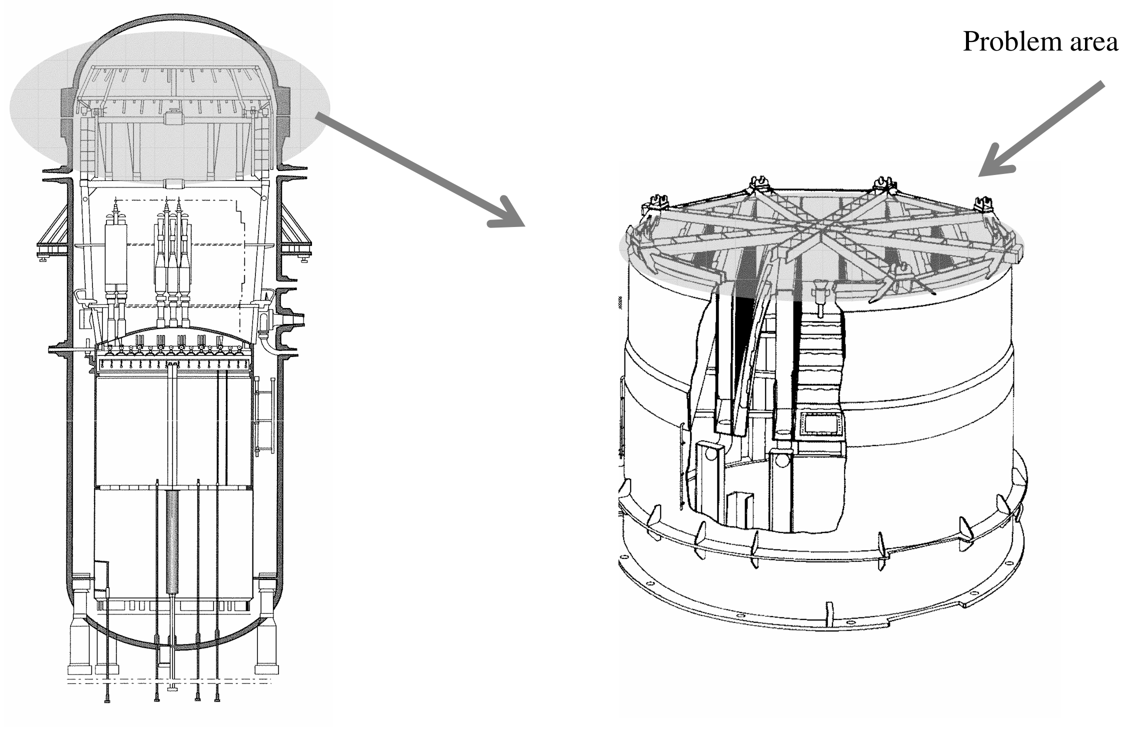
# "ALARA" VERSUS REACTOR SAFETY CONCERN – A PRACTICAL CASE

**S. Hennigor, B. Ögren**
Forsmarks Kraftgrupp AB, Sweden

## Introduction

During the outage 2003 at Forsmark unit 2 it was planned to make a modification to the moist separator (an internal part of the reactor vessel), see Figure 1. The work was initiated due to extensive cracking found in welds, which challenged the mechanical integrity of the moist separator and also loose parts lost in the reactor vessel. The cracks had been known for several years but until now no measures were deemed necessary.

**Figure 1. The problem area on the upper part of the steam dryer**

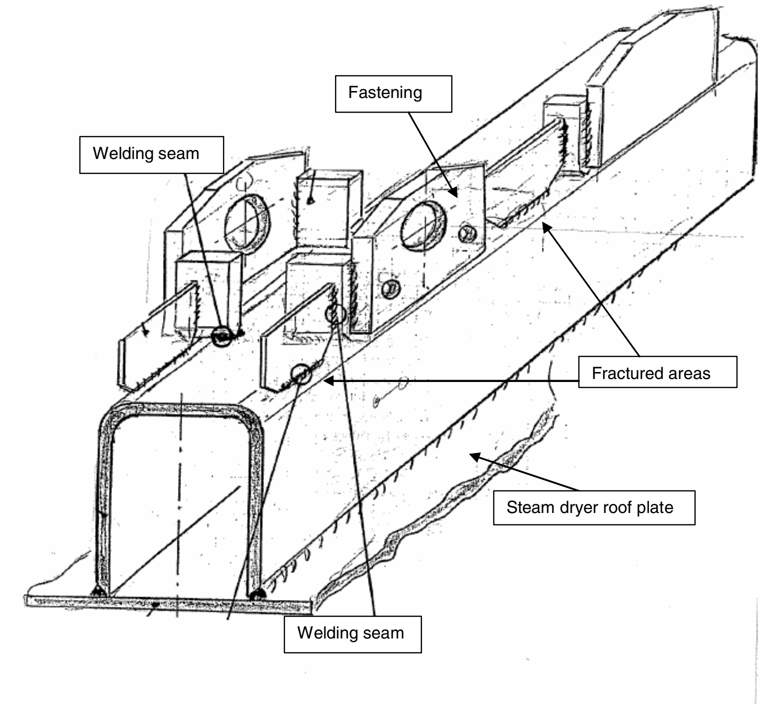


## Description of the problem

Just before taking Forsmark unit 2 into operation 1980 it was decided to provide additional plates on top of the steam dryer in order to enhance the performance. However, due to operating experience from the Finnish site TVO (which has two reactors similar to Forsmark 2), the plates were not installed. The consoles supporting the plates were left in place, assuming that it was without risk to leave them as it was.

After some years of operation cracks were observed in the weldings connecting the supporting consoles to the beams of the steam dryer, see Figure 2.

**Figure 2. Consoles welded on top of steam dryer**




Not until the beginning of year 2003 the cracks were judged to be a serious challenge to the mechanical integrity of the reactor vessel internals (integrity of the beams) and also the risk of having parts coming loose in the reactor vessel was of concern.

The decision was taken to take measures on the above already during the outage year 2003, only six months ahead in time. One of the reasons was that the outages during 2004-2005 only are scheduled for 9 days, which is to short for this kind of work to be performed.

Measures to be performed were to remove the supporting consoles and to weld an extra beam on to the existing, thus ensuring that the cracks and fractures in the original beams would not endanger the integrity of the structure. Also there will be no need for future materials inspection of the old beams since their function is entirely replaced by the new ones.

**Planning of the work**

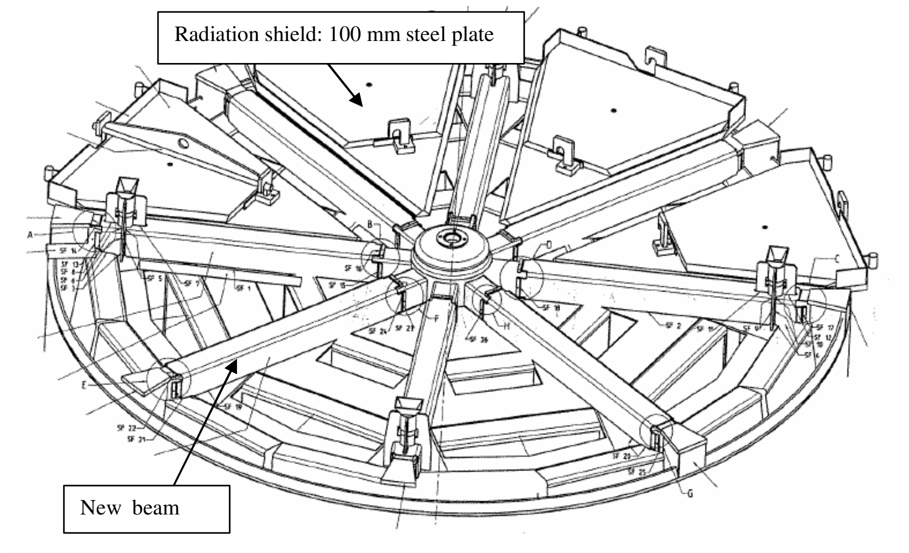
In order to perform the acquired operations it was necessary to expose the upper part of the steam dryer from the water pool.

The short planning time, approximately six months imposed some difficulties:

- since the work was not known to be performed no dose rate measurements with exposed upper part of the steam dryer were performed during the outage year 2002;
- to short time to develop and implement automatic cutting, grinding and welding machines;
- limited time to optimise cutting, grinding and welding methods (based on more or less manual work methods).

Of course a critical factor was to determine the dose rate at the working place on top of the steam dryer. This had to be done based on measurements performed at the steam dryer of Forsmark 3 and measurements performed at Forsmark 2 during other previous work on internals. Also additional shielding calculations were performed. Dose rate targets were set to 0.5 mSv/h in general above shielded parts of the working area but higher near the unshielded beams. According to measurements and calculations a radiation shield consisting of steel plates forming a working platform were constructed, see Figure 3. The steel plates were to be supplemented by additional lead blankets over the beams not being worked on.

**Figure 3. Steel plate radiation shields**




Another problem related to radiation protection was how to minimise the spread of air and surface contamination, especially from the cutting and grinding activities. A related problem would be heat load to workers if they had to work in heavy protection clothing and within a tent. To put focus on the radiological and industrial risks involved a comprehensive risk assessment were performed and distributed to all parties involved, including the contractor chosen for the performance of work. In perspective this was a very effective way to put these questions on the table and acknowledged by everyone involved.

Early dose predictions showed a total collective dose of 250 man.mSv and with individual doses near 20 mSv for several persons. Can this be justified according to the ALARA principle? The immediate answer would be no, since enough time had not been given to proper acquire dose rate data based on measurements, develop and implement suitable working methods and tools and to optimise the protection measures accordingly. There is also an obvious risk that due to dose limitations, and other radiological matters, interruption of the work may be necessary, thus leaving the steam dryer in an even worse condition than before.

Due to the judgment that continued operation without the fixing of the steam dryer, the work planning was continued and the work targeted to be performed as planned during the outage. However, the Forsmark Safety Committee, which has to approve all safety related work on the reactors, required that a set of check points were established. At this check points continued work should be reviewed and evaluated. The evaluation included participation from the Unit Manager and the Radiation Protection Manager.

Check points established were:

1. factory Acceptance Test verifying the contractors working methods and tools: check for labor time required;
2. lifting of reactor vessel head: check for source term accuracy;
3. when the shielding steel plates were in place: check the actual dose rate compared with targets set up;
4. during work performance: continuously check collective dose and individual doses. Constraints were set at the maximum collective dose of 350 man.mSv and maximum individual dose of 12 mSv.

The continued planning, together with the contractor, resulted in the following major improvements regarding radiological safety:

- cutting the consoles by saw blade operated at slow speed and using a semiautomatic method minimising manual operations;
- slow speed grinding method;
- choice of speedy welding method (only manual welding was possible), cutting work time for welding by 2/3 compared to the initial method proposed;
- training and verification performed at a mock-up built for the purpose;
- special information given to the workers involved regarding radiological and industrial safety matters for this particular work.

The choice of working methods made it possible to perform the work without tents over the work place, but extra ventilation and thorough cleaning procedures had to be implemented.

An adjusted dose prediction based on this planning and the verification of work methods and tools (check point 1) resulted in a anticipated collective dose of 160-180 man.mSv, with no individual dose in excess of 12 mSv.

### *Performance of work*

Measurements of source term and dose rates with shielding plates in place (check points 2 and 3) showed dose rates on working area in excess of predicted values. General dose rates were in the range of 0,7-0,8 mSv/h with values of about 2 mSv/h on working distance from unshielded beams.

New dose predictions based on this data showed that the work could not be performed within the given constraint given by check point 4. An alternative to the original plan was worked out resulting in only modify 4 of 8 beams during this outage. The 4 “worst” beams were chosen and stress calculations showed that the integrity of the structure would be enough to justify continued operation, even if 4 beams were left unmodified.

The actual work was performed according to the modified plan without any complications, incidents or accidents. Daily meetings were held between contractor’s staff, operational staff and radiation protection staff. During these meetings the resulting doses were closely monitored and communicated to everyone involved.

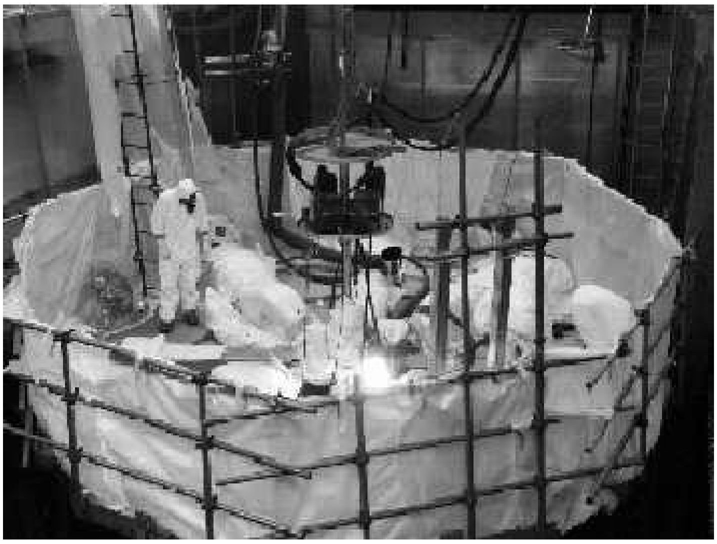
The total collective dose resulting was measured to be 165.5 man.mSv and the maximum individual dose 10.3 mSv (welder). We considered this to be an acceptable output, taking into account the limited time for optimisation and lack of source term data. The involvement of a highly professional and engaged contractor at an early stage greatly promoted the output. But 4 beams still have to be modified in the forthcoming 2-3 years.

## Conclusions and lessons learned

- Work of this kind should not be performed without proper time allocated for the collection of realistic measurement data and optimisation of working methods, tools and protective measures. At least 12 months seems appropriate.
- Independent shielding calculations should be performed especially on complex geometries. In this case an underestimation of the source term was done, resulting in higher dose rates measured at the working area than calculated.
- Making a formal and comprehensive risk assessment will place focus also on these matters and not only on the technical engineering work. It also provides an excellent base for communication between all involved parties, including contractors.
- Dedicated daily follow-up meetings proved to be of great value in coordinating the work and communicate actual dose figures among all involved.
- Pre mock-up training and additional dedicated information given to the personal involved proved to be of greatest importance. This greatly enhances workers involvement and awareness, besides enhancing the quality of work performed.

- Establishing well defined check points, including alternative actions, before actually starting the work minimises the risk of having negative surprises which could lead to loss of outage time and poor quality of work.

**Figure 6. Working area with protective measures implemented**



# RECENT INTERNATIONAL DEVELOPMENTS ON CONTAMINATION LIMITS ON PACKAGES

**J. Hesse**
RWE Power, Germany

**B. Lorenz**
GNS, Germany

## Introduction

In 1998 several events had been reported in the German media on non-compliance with the contamination limits for transport of spent fuel to and from the reprocessing facilities in France and the United Kingdom. The reporting developed to a tremendous media campaign and led to a transport stop announced by the authorities in several European countries, e.g. in Germany the transport of spent fuel from power reactors was interrupted for about 3 years.

In the meantime a lot of efforts were taken by the industry and the authorities to overcome the situation and to agree on a new concept of contamination prevention and control. Today it can be stated, that these measures have proven to be effective and no non-compliance with the contamination limits have been observed for spent fuel transports.

It soon became evident that none of the reported contamination findings led to any remarkable dose to members of the public or to any radiation worker involved in the transports. So from the very beginning of that discussion many parties involved in the affair were convinced that the existing system of contamination limits in the transport regulations was no longer up-to-date and needed modernisation.

A number of proposals were made by different countries and on behalf of the nuclear industry by the WNTI World Nuclear Transport Institute. Finally, the IAEA launched a Co-ordinated Research Project (CRP) on the Radiological Aspects of Package and Conveyance Non-Fixed Contamination to deal with all items of concern.

One of the major tasks of the CRP, which lasted from 2001 till 2003, was to develop a new model for contamination limits for the transport of radioactive material and associated equipment. WNTI was one of the 7 participating parties, including also France, Germany, Japan, Sweden, United Kingdom and United States. The WNTI working group was formed basically by the German organisation VGB, which comprises all German nuclear power plants.

This WNTI working group has created a model on its own as input to the CRP. In several special meetings with the other groups it was developed to an international radiological model, and was then adopted by the whole CRP group.

In the following we will describe the radiological model proposed to calculate doses from non-fixed (removable) surface contamination during transport of radioactive material and the results of the model calculations. Taking into account these results and also the efforts necessary for decontamination and contamination control we finally will discuss, whether the actual contamination limits are still appropriate.

## Fairbairne model

The starting point of the considerations was the analysis of the Fairbairn model, which up to now has served as the basis for the current surface contamination values for transport. The results of this old model, described in a paper of 1961, have since then been in use (having undergone the only change in the transition from Curie to Becquerel as the new unit for radioactivity, rounding up from 3.7 to 4).

The Fairbairn model is based on a single exposure situation involving;

- “most hazardous radioisotopes in common use”: Pu 239, Ra 226, Sr 90;
- “very dusty operations” with a resuspension factor of $4 \cdot 10^{-5}$ $m^{-1}$;
- 2 000 hours per year working in that “dusty” atmosphere;
- taking into account skin contamination and inhalation only;
- 50 mSv/a as basis for deriving the contamination limits;
- no considerations of the doses of members of the public.

This approach does not take into account the very large differences in the radiological properties of radionuclides and is not appropriate for the real transport situations nowadays. So the Fairbairn model can be judged as very conservative. On the flipside it derives the contamination limits from the basic individual dose limit of workers, what probably would no longer be accepted nowadays.

## New model

After an analysis of the Fairbairne model the WNTI/VGB group decided to built up a new model taking into account

- radionuclide specific data of all radionuclides;
- different kinds of packages;
- the single steps during a transport (time of steps, distance between worker and package etc);
- differentiation between indoor and outdoor operations;

- all possible exposure pathways;
- the exposure of workers and members of the public as well.

### *Types of packages*

As a reasonable compromise between all existing packages and the objective of the model to be representative for any type of packages, we chose four different package types as representatives: Small manually handled packages, small remotely handled packages (200 l drum), large remotely handled packages (20' container), and fuel flasks. An overview is given in the following Table.

**Table 1. Overview of package types**

| Package/container type | Dimensions | Total volume | Total surface area |
|---|---|---|---|
| SM: small manual (parcel) | 0.3 x 0.3 x 0.3 $m^3$ | 0.03 $m^3$ | 0.5 $m^2$ |
| SR: small remote (200 l drum) | height 0.9 m, diameter 0.6 m | 0.25 $m^3$ | 2.3 $m^2$ |
| LR: large remote (20' container) | 5.9 x 2.4 x 2.4 $m^3$ | 34 $m^3$ | 68 $m^2$ |
| FF: fuel flask (with fins) | length 6 m, diameter 2.5 m | 29 $m^3$ | 130 $m^2$ (incl. fins) |

### Steps during transport

The basic modelling principle was the assumption that any transport can be defined as a sequence of steps and several actions (sub-steps) during these steps. Table 2 gives an overview of the main steps, the actions they consists of and the groups of persons who might be possibly exposed.

The transport process which is modelled here covers the period from the final inspection of the package before detachment up to the receiving inspection and transfer to the final destination place. The starting point of the model is within a nuclear facility. It was thought to be adequate to start at that moment, when a package is decided to become a package for transport. There might be several processes in a nuclear facility, where radioactive material is handled and will remain within the facility. All these processes are under the supervision of the radiation protection regime for the facility, basically governed by the principles of the IAEA Basic Safety Standards and the corresponding national regulations, probably ruled by a license. It is therefore not necessary to cover these actions by the transport model. Thus the model does not cover any preparatory steps like e.g. decontamination of the package or container to the contamination limits.

The endpoint of the model is the receiving inspection and the end control of the transport equipment, especially the vehicle. It does not cover the opening of the package and removal of the radioactive contents as this will be the beginning of a new action within the framework of the site/facility license.

**Workers assigned to the actions**

The model also takes into account that the different steps of a transport are done by different workers. That is shown in Table 2, too, as follows:

- Workers A and B are involved in the package preparation and the transfer to the conveyance. In the case of small manually handled packages it is assumed that there is no special transfer worker B.
- Worker C is the driver.
- Workers F, G, H are working at the transfer site, T and U at the consignee's premises.
- Multiple letters per box indicate that the step may be carried out by either of the persons.

**Table 2. Steps relevant for the transport model, persons involved**

| Main step | Action | Persons involved | Workers | | | |
|---|---|---|---|---|---|---|
| | | | SM | SR | LR | FF |
| 1. Final Inspection of Package | 1.1 Visual inspection | Personnel | A | A | A | A |
| | 1.2 Dose rate meas. | | A | A | A | A |
| | 1.3 Contamination measurement (final meas.) | | A | A | A | A |
| | 1.4 Labelling of package | | A | A | A | A |
| 2. Loading onto conveyance | 2.1 Transfer from site to conveyance | Personnel | C | BC | BC | B |
| | 2.2 Fastening, loading, lifting and fixing | | C | BC | BC | B |
| | 2.3 Dose rate meas. at conveyance | | AC | AC | A | A |
| | 2.4 Contamination meas. of conveyance | | AC | AC | A | A |
| | 2.5 Placarding of conveyance | | AC | AC | A | A |
| 3. Movement phase | 3.1 Movement (with packages) | Personnel | C | C | C | C |
| | 3.2 Unforeseen interruptions | | C | C | C | C |
| | 3.1a Movement, public, road/rail | Public | no | no | no | no |
| | 3.1b Movement, public, air | | no | no | no | no |
| | 3.1c Movement, public, sea | | no | no | no | no |
| | 3.3 Regular stops | | no | no | no | no |
| 4. Transfers during transport | 4.1 Unloading (incl. sub-steps) from conv. #1 | | | | | |
| | 4.1.1 dose rate & contam. meas. | Personnel, Public | no | F | H | H |
| | 4.1.2 unfixing, fastening, lifting | | F | F | FG | FG |
| | 4.2 Loading (incl. sub-steps) on conv. #2 | | | | | |
| | 4.2.1 transfer, loading, fixing | | F | F | FG | FG |
| | 4.2.2 dose rate meas. at conveyance | | F | F | H | H |
| | 4.2.3 contamination meas. of conveyance | | F | F | H | H |
| | 4.2.4 placarding of conveyance | | F | F | F | F |
| | 4.3 Regular stops | Public | no | no | no | no |
| 5. Receiving inspection and unloading | 5.1 Visual inspection of load | Personnel | T | T | T | T |
| | 5.2 Dose rate meas. conveyance | | no | no | no | T |
| | 5.3 Unfixing, fastening, lifting, unloading | | C | CU | CU | U |
| | 5.4 Transfer from conveyance to consignee | | C | CU | CU | U |
| | 5.5 Dose rate measurement package | | T | T | T | T |
| | 5.6 Contamination meas. package | | no | T | T | T |
| | 5.7 Contamination meas. empty conveyance | | no | no | T | T |

That gives a general idea of what each worker group is meant to consist of. There might be deviations from the actual working conditions, but the model is still realistic and also sufficiently conservative, which requires slight overestimation rather than underestimation of working time and tasks per person.

### Annual exposure time

For all steps the annual time of exposure was modelled by multiplying the time of the single step by the number of such steps a person might do per year as shown in the following table.

Table 3. **Parameters concerning shipments and annual working time**

| Parameter | unit | Parameter value for: | | | | |
|---|---|---|---|---|---|---|
| | | SM | | SR | LR | FF |
| number of days per year | d/a | 250 | | 250 | 250 | 250 |
| working hours per day | h/d | 8 | | 8 | 8 | 8 |
| annual working time | h/a | 2 000 | | 2 000 | 2 000 | 2 000 |
| cut-off annual working time | [-] | 75% | | 75% | 75% | 75% |
| number of loads handled per day | 1/d | 2* | 1* | 1 | 1 | 0.5 |
| packages per conveyance | [-] | 25* | 100* | 42 (25)** | 1 | 1 |
| packages per year | 1/a | 25 000 | | 6 250 | 250 | 125 |
| geometry of packages on conveyance | | 2 layers of 15/10 packages | | 2 layers of 3x7 drums | 1 container | 1 fuel flask |

* Either 1 load of 100 packages or 2 loads of 25 packages per day – differences in the type of conveyance are assumed.
** It is assumed that a load consists of 42 packages of which 25 are handled by a single worker.

The next table shows as an example the assumed time and some other parameter details for the step "visual inspection".

Table 4. **Parameters for the substep "visual inspection"**

| 1.1 Visual inspection | | | SM | SR | LR | FF |
|---|---|---|---|---|---|---|
| exposure time per package | $t_{exp,s}$ | min | 0,5 | 1 | 5 | 10 |
| exposure time per year | $t_{exp,total}$ | h/a | 208 | 175 | 21 | 21 |
| persons involved | | | A | A | A | A |
| distance to package | $l_{extirr}$ | m | 1 | 1 | 1 | 1 |
| number of packages | | | 1 | 1 | 1 | 1 |
| exposure geometry | | | 24 | 14 | 20 | 20 |
| resuspension rate | $f_{resus}$ | 1/h | 1,00E-04 | 1,00E-04 | 1,00E-04 | 1,00E-04 |
| act. conc. air (room) 1 Bq/cm², 1 package | $A_{air,room}$ | Bq/m³ | 1,08E-03 | 4,52E-03 | 1,70E-02 | 3,25E-02 |

### Exposure pathways

The following exposure pathways pertaining to non-fixed surface contamination on packages are included in the model:

- external irradiation from the removable surface contamination (not by the contents of the package);
- inhalation of radioactive aerosols re-suspended from the contaminated surface;
- ingestion of radioactivity via a hand-to-mouth pathway, i.e. the hand touches the radio-actively contaminated surface and subsequently gets into contact with the mouth;
- skin contamination resulting from direct skin contact with the contaminated surface.

In addition, there are other pathways which, however, may give rise only to insignificant dose contributions. They have been considered for a number of scenarios especially in connection with a potential dose to members of the public but have been disregarded for further inclusion in the model. Examples of such insignificant pathways are:

- ingestion of foodstuff grown in the vicinity of a transport path;
- external radiation or other dose paths from fall-out or wash-out of radioactivity from contaminated packages;
- secondary re-suspended activity from deposited surface contamination;
- contamination build-up by frequent and continued use of places/ways for transport or temporary storage of packages;
- secondary contamination of passenger areas through ventilation connections with freight areas containing packages with radioactive materials.

For the sake of time we cannot show all the parameters of the model here. Parameters had to be chosen also for all the different exposure pathways, e.g. for the volume of a room with packages, for the detailed geometry to calculate the exposure by direct radiation in all different cases, the dose coefficients, etc.

All parameters were discussed several times. However, in the CRP consensus between all groups could be found concerning all parameters of the model.

### Calculations and results

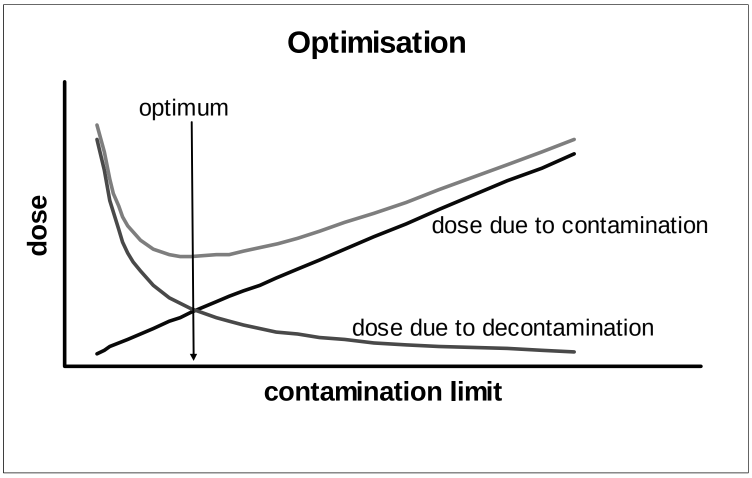
Calculations were made nuclide specifically by assuming a contamination of 1 Bq/cm² on the entire surface of every package. Results were calculated in Sv/(Bq/cm²) for every single step and every single involved person. For an easier comparison with the existing contamination limit we calculated in a second step the contamination leading to a dose of 2 mSv/a for the most exposed worker and the contamination leading to a dose of 0.3 mSv/a for a member of the public. A part of the results are shown in Table 5. The complete table of all radionuclides can be seen in the Annex.

**Table 5. Final results (part of the complete table) for surface contamination levels in Bq/cm², which correspond to a dose constraint of 2 mSv/a for workers and of 0.3 mSv/a for members of the public**

| | Derived Level Workers [Bq/cm²] | | | | Derived Level Public [Bq/cm²] | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Nuclide | W-SM | W-SR | W-LR | W-FF | P-SM | P-SR | P-LR | P-FF | Overall Min. |
| Cm-248 | 0,1 | 1,6 | 2,1 | 1,7 | 31 | 6 | 68 | 46 | 0,1 |
| Co-55 | 135 | 134 | 305 | 407 | 1,8E+5 | 1,9E+4 | 4,0E+3 | 7,6E+3 | 134 |
| Co-56 | 81 | 80 | 185 | 249 | 9,7E+4 | 1,1E+4 | 2,3E+3 | 4,5E+3 | 80 |
| Co-57 | 2,3E+3 | 2,2E+3 | 4,9E+3 | 6,4E+3 | 2,3E+6 | 2,8E+5 | 6,8E+4 | 1,3E+5 | 2,2E+3 |
| Co-58 | 270 | 268 | 621 | 834 | 3,2E+5 | 3,6E+4 | 7,9E+3 | 1,5E+4 | 268 |
| Co-58m | 1,2E+5 | 1,3E+5 | 1,3E+5 | 1,3E+5 | 3,5E+8 | 7,1E+7 | 3,1E+8 | 3,5E+8 | 1,2E+5 |
| Co-60 | 109 | 108 | 245 | 323 | 1,1E+5 | 1,3E+4 | 3,2E+3 | 6,1E+3 | 108 |
| Cr-51 | 8,2E+3 | 8,2E+3 | 1,8E+4 | 2,4E+4 | 1,0E+7 | 1,1E+6 | 2,4E+5 | 4,7E+5 | 8,2E+3 |
| Cs-129 | 893 | 886 | 2,1E+3 | 2,8E+3 | 1,2E+6 | 1,2E+5 | 2,6E+4 | 4,9E+4 | 886 |
| Cs-131 | 7,7E+3 | 7,7E+3 | 1,7E+4 | 2,2E+4 | 1,0E+7 | 1,1E+6 | 2,4E+5 | 4,6E+5 | 7,7E+3 |
| Cs-132 | 362 | 359 | 836 | 1,1E+3 | 4,7E+5 | 5,0E+4 | 1,0E+4 | 2,0E+4 | 359 |
| Cs-134 | 129 | 128 | 227 | 266 | 1,7E+5 | 2,0E+4 | 4,9E+3 | 9,3E+3 | 128 |
| Cs-134m | 6,4E+3 | 6,4E+3 | 1,1E+4 | 1,2E+4 | 1,2E+7 | 1,2E+6 | 2,6E+5 | 5,0E+5 | 6,4E+3 |
| Cs-135 | 4,0E+3 | 4,6E+3 | 4,6E+3 | 4,6E+3 | 6,7E+6 | 1,4E+6 | 1,5E+7 | 1,0E+7 | 4,0E+3 |
| Cs-136 | 122 | 121 | 274 | 363 | 1,6E+5 | 1,7E+4 | 3,6E+3 | 6,9E+3 | 121 |
| Cs-137 | 284 | 283 | 439 | 487 | 3,8E+5 | 5,0E+4 | 1,3E+4 | 2,5E+4 | 283 |
| Cu-64 | 1,3E+3 | 1,3E+3 | 2,8E+3 | 3,6E+3 | 1,8E+6 | 1,9E+5 | 4,0E+4 | 7,6E+4 | 1,3E+3 |

## Discussion and conclusions

The model calculations have been internation-ally harmonised within the IAEA CRP for all necessary parameters. The CRP failed however to agree finally about the mode of deriving a contamination limit or level. The discussion on this subject was rather divers and no consensus could be achieved.




However, it is essential for radio protection to find and establish an adequate level of contamination. It has to be taken into account that too low contamination limits can have an effect converse to the wished effect of protection.

The lower the contamination limits are the higher efforts of decontamination and measurements are needed to ensure compliance with the limits. This work has to be done in the radiation field produced by the contents of the packages. So it results in a higher dose of the workers. This situation is shown in Figure 1. The dose due to the contamination competes with the dose due to decontamination. If the legal contamination limit is higher than the optimum, it is possible to reach the optimum by putting a lower “internal” limit. However, if the legal contamination limit is lower than this optimum, it is impossible to reach the optimum. This is the situation we have to get along with today.

The following tables show for the two radionuclides $^{60}$Co and $^{137}$Cs, which most times dominate the contamination from a NPP, the doses corresponding with the actual contamination limit of 4 Bq/cm².

**Table 6. Annual effective dose of the most exposed workers due to a contamination of 4 Bq/cm²**

| | Small manual | Small remote | Large remote | Fuel flask |
|---|---|---|---|---|
| $^{60}$Co | 73 µSv/a | 74 µSv/a | 33 µSv/a | 25 µSv/a |
| $^{137}$Cs | 28 µSv/a | 28 µSv/a | 18 µSv/a | 16 µSv/a |

**Table 7. Annual effective dose of a member of the public due to a contamination of 4 Bq/cm²**

| | Small manual | Small remote | Large remote | Fuel flask |
|---|---|---|---|---|
| $^{60}$Co | 0,011 µSv/a | 0,092 µSv/a | 0,375 µSv/a | 0,197 µSv/a |
| $^{137}$Cs | 0,003 µSv/a | 0,024 µSv/a | 0,092 µSv/a | 0,048 µSv/a |

These doses are calculated still with the very conservative assumption that the complete surface of every package is contaminated up to the actual contamination limit of 4 Bq/cm², which means a factor of conservaty in the order of 10 or even 100. In the case of fuel flasks these values are based on the assumption that one worker handles 125 flasks per year. So the dose per flask is only 0.2 µSv/flask ($^{60}$Co), respectively 0.1 µSv/flask ($^{137}$Cs). These calculated potential doses should be compared with the real doses workers get to reach sure compliance with the contamination limits. Referring to a paper of J.P. Degrange *et al* from the last European ISOE workshop in 2002, the operations of prevention, elimination and monitoring of the surface contamination of the irradiated fuel casks before shipment contribute significantly with about 42% to the collective dose of the involved workers. In total they estimated as sum for all French NPPs a dose of 1.3 manSv per year for these steps before the actual shipment. From German NPPs we know that some workers get doses in the order of 1mSv due to the decontamination and monitoring of one single cask.

Additionally the calculations show that the dose per unity surface activity (Sv/a per Bq/cm²) strongly depends on the radionuclide. The calculated values reach over about seven orders of magnitude. This is due to the fact that the different radiological importance of the different radionuclides directly influences the calculation. Today only one order of magnitude is reflecting the differences between radionuclides. The actual limitation by only two values (0.4 Bq/cm² for alpha emitters and 4 Bq/cm² for the other nuclides) does not approximately reflect the radiological importance of the different radionuclides adequately.

When setting contamination limits there are of course also other aspects to be taken into account as:

- a reached level of cleanliness should not be given up without reason;
- the dose due to contamination should be only a part of the dose for the whole process;
- the contamination limits for transports should be in compliance with the contamination limits in the receiving facilities;
- new contamination limits must be justifiable also in a political debate with the public.

Even taking into account these additional conditions it seems to be appropriate to put the contamination limits on the basis of the described new international model of the IAEA CRP. It is much more realistic than the Fairbairne model of 1961 and still rather conservative. This would mean to substitute the actual limits by radionuclidspecific ones and to consider an appropriate constraint, e.g. 2 mSv/a for workers, for the derivation of these limits.

***Annex 1.*** **Final Results for Surface Contamination Levels in Bq/Cm$^2$ Which Correspond To a Dose Constraint of 2 Msv/A For Workers and of 0.3 Msv/A For Members of the Public**

| | Derived Level Workers [Bq/cm$^2$] | | | | Derived Level Public [Bq/cm$^2$] | | | | |
|---|---|---|---|---|---|---|---|---|---|
| **Nuclide** | W-SM | W-SR | W-LR | W-FF | P-SM | P-SR | P-LR | P-FF | **Overall Min.** |
| Ac-225 | 21 | 158 | 190 | 171 | 5,1E+3 | 1,0E+3 | 8,9E+3 | 7,0E+3 | 21 |
| Ac-227 | 0,036 | 0,4 | 0,6 | 0,5 | 8 | 1,7 | 18 | 12 | 0,036 |
| Ac-228 | 204 | 223 | 400 | 454 | 1,3E+5 | 2,0E+4 | 8,4E+3 | 1,6E+4 | 204 |
| Ag-105 | 405 | 402 | 750 | 901 | 6,0E+5 | 6,6E+4 | 1,4E+4 | 2,8E+4 | 402 |
| Ag-108m | 149 | 155 | 354 | 447 | 7,9E+4 | 1,2E+4 | 4,6E+3 | 8,5E+3 | 149 |
| Ag-110m | 97 | 97 | 223 | 296 | 9,8E+4 | 1,2E+4 | 2,9E+3 | 5,4E+3 | 97 |
| Ag-111 | 2,4E+3 | 2,8E+3 | 3,4E+3 | 3,4E+3 | 2,3E+6 | 4,0E+5 | 2,8E+5 | 4,9E+5 | 2,4E+3 |
| Al-26 | 102 | 102 | 225 | 290 | 8,9E+4 | 1,1E+4 | 3,1E+3 | 5,9E+3 | 102 |
| Am-241 | 0,5 | 6 | 8 | 6 | 111 | 23 | 244 | 165 | 0,5 |
| Am-242m | 0,6 | 6 | 8 | 7 | 126 | 26 | 277 | 187 | 0,6 |
| Am-243 | 0,5 | 6 | 8 | 6 | 113 | 23 | 249 | 169 | 0,5 |
| As-72 | 143 | 142 | 312 | 408 | 1,9E+5 | 2,1E+4 | 4,3E+3 | 8,3E+3 | 142 |
| As-73 | 8,8E+3 | 1,3E+4 | 1,6E+4 | 1,7E+4 | 4,3E+6 | 8,2E+5 | 1,1E+6 | 1,8E+6 | 8,8E+3 |
| As-74 | 320 | 318 | 691 | 891 | 3,8E+5 | 4,3E+4 | 9,8E+3 | 1,9E+4 | 318 |
| As-76 | 490 | 488 | 887 | 1,1E+3 | 7,3E+5 | 8,2E+4 | 1,8E+4 | 3,5E+4 | 488 |
| As-77 | 5,7E+3 | 6,1E+3 | 6,9E+3 | 7,0E+3 | 9,3E+6 | 1,6E+6 | 8,7E+5 | 1,6E+6 | 5,7E+3 |
| At-211 | 162 | 611 | 661 | 636 | 4,2E+4 | 8,5E+3 | 6,3E+4 | 5,4E+4 | 162 |
| Au-193 | 1,7E+3 | 1,7E+3 | 3,4E+3 | 4,4E+3 | 2,3E+6 | 2,5E+5 | 5,3E+4 | 1,0E+5 | 1,7E+3 |
| Au-194 | 264 | 261 | 613 | 832 | 3,4E+5 | 3,7E+4 | 7,6E+3 | 1,5E+4 | 261 |
| Au-195 | 2,9E+3 | 2,9E+3 | 5,9E+3 | 7,2E+3 | 2,2E+6 | 3,0E+5 | 9,5E+4 | 1,8E+5 | 2,9E+3 |
| Au-198 | 477 | 474 | 820 | 954 | 7,5E+5 | 8,5E+4 | 1,9E+4 | 3,6E+4 | 474 |
| Au-198m | 232 | 231 | 445 | 542 | 3,1E+5 | 3,5E+4 | 8,0E+3 | 1,5E+4 | 231 |
| Au-199 | 2,4E+3 | 2,4E+3 | 4,3E+3 | 5,0E+3 | 2,6E+6 | 3,4E+5 | 9,2E+4 | 1,7E+5 | 2,4E+3 |
| Ba-131 | 506 | 502 | 1,2E+3 | 1,5E+3 | 6,6E+5 | 7,1E+4 | 1,5E+4 | 2,8E+4 | 502 |
| Ba-133 | 611 | 607 | 1,3E+3 | 1,7E+3 | 6,8E+5 | 8,0E+4 | 1,9E+4 | 3,6E+4 | 607 |
| Ba-133m | 1,7E+3 | 1,7E+3 | 2,3E+3 | 2,5E+3 | 4,1E+6 | 4,6E+5 | 1,0E+5 | 2,0E+5 | 1,7E+3 |
| Ba-140 | 109 | 108 | 240 | 314 | 1,4E+5 | 1,5E+4 | 3,3E+3 | 6,3E+3 | 108 |
| Be-7 | 5,2E+3 | 5,2E+3 | 1,2E+4 | 1,7E+4 | 6,5E+6 | 7,1E+5 | 1,5E+5 | 2,9E+5 | 5,2E+3 |
| Be-10 | 1,21E+3 | 2,4E+3 | 2,5E+3 | 2,5E+3 | 4,8E+5 | 9,9E+4 | 1,1E+6 | 7,2E+5 | 1,21E+3 |
| Bi-205 | 175 | 173 | 406 | 550 | 2,2E+5 | 2,4E+4 | 5,0E+3 | 9,7E+3 | 173 |
| Bi-206 | 83,4 | 83 | 193 | 260 | 1,1E+5 | 1,1E+4 | 2,4E+3 | 4,6E+3 | 82,7 |
| Bi-207 | 174,0 | 173 | 397 | 527 | 1,8E+5 | 2,2E+4 | 5,1E+3 | 9,7E+3 | 172,9 |
| Bi-210 | 224,27 | 1,6E+3 | 1,9E+3 | 1,7E+3 | 5,0E+4 | 1,0E+4 | 1,1E+5 | 7,5E+4 | 224,27 |
| Bi-210m | 6,39E+00 | 6,99E+01 | 9,30E+01 | 7,76E+01 | 1,4E+3 | 278 | 2,7E+3 | 2,0E+3 | 6,39E+00 |
| Bi-212 | 194,0 | 210 | 477 | 594 | 9,9E+4 | 1,5E+4 | 6,3E+3 | 1,2E+4 | 194 |
| Bk-247 | 0,3 | 3,4 | 5 | 3,7 | 67 | 14 | 148 | 100 | 0,3 |
| Bk-249 | 132 | 1,4E+3 | 1,9E+3 | 1,6E+3 | 2,9E+4 | 5,9E+3 | 6,4E+4 | 4,3E+4 | 132 |
| Br-76 | 110 | 109 | 248 | 330 | 1,5E+5 | 1,6E+4 | 3,2E+3 | 6,2E+3 | 109 |
| Br-77 | 844 | 836 | 2,0E+3 | 2,7E+3 | 1,1E+6 | 1,2E+5 | 2,4E+4 | 4,7E+4 | 836 |
| Br-82 | 103 | 102 | 240 | 327 | 1,3E+5 | 1,4E+4 | 3,0E+3 | 5,7E+3 | 102 |
| C-11 | 260,2 | 258 | 618 | 852 | 3,4E+5 | 3,6E+4 | 7,4E+3 | 1,4E+4 | 258,0 |
| C-14 | 1,01E+4 | 1,4E+4 | 1,4E+4 | 1,4E+4 | 2,3E+6 | 4,7E+5 | 5,1E+6 | 3,5E+6 | 1,01E+4 |
| Ca-41 | 2,63E+4 | 3,3E+4 | 3,3E+4 | 3,3E+4 | 4,9E+7 | 1,0E+7 | 1,1E+8 | 7,3E+7 | 2,63E+4 |
| Ca-45 | 4,3E+3 | 9,2E+3 | 9,5E+3 | 9,3E+3 | 1,7E+6 | 3,5E+5 | 3,8E+6 | 2,6E+6 | 4,3E+3 |
| Ca-47 | 227,0 | 225 | 488 | 629 | 2,7E+5 | 3,1E+4 | 7,0E+3 | 1,3E+4 | 225,4 |
| Cd-109 | 1,16E+3 | 1,7E+3 | 1,9E+3 | 2,0E+3 | 6,6E+5 | 1,3E+5 | 1,9E+5 | 3,0E+5 | 1,16E+3 |
| Cd-113m | 198 | 362 | 369 | 365 | 8,9E+4 | 1,8E+4 | 2,0E+5 | 1,3E+5 | 198 |
| Cd-115 | 603 | 599 | 1,2E+3 | 1,4E+3 | 7,7E+5 | 8,9E+4 | 2,0E+4 | 3,9E+4 | 599 |
| Cd-115m | 1,1E+3 | 1,6E+3 | 1,8E+3 | 1,8E+3 | 7,2E+5 | 1,4E+5 | 3,0E+5 | 4,3E+5 | 1,1E+3 |
| Ce-139 | 1,5E+3 | 1,5E+3 | 3,4E+3 | 4,3E+3 | 1,2E+6 | 1,6E+5 | 4,7E+4 | 8,9E+4 | 1,5E+3 |
| Ce-141 | 2,0E+3 | 2,4E+3 | 4,0E+3 | 4,4E+3 | 1,1E+6 | 1,9E+5 | 1,0E+5 | 1,8E+5 | 2,0E+3 |
| Ce-143 | 767 | 763 | 1,4E+3 | 1,7E+3 | 1,0E+6 | 1,2E+5 | 2,7E+4 | 5,2E+4 | 763 |
| Ce-144 | 382 | 965 | 1,1E+3 | 1,0E+3 | 1,3E+5 | 2,5E+4 | 1,0E+5 | 1,2E+5 | 382 |
| Cf-248 | 2,4 | 28 | 38 | 31 | 529 | 107 | 1,2E+3 | 788 | 2,4 |
| Cf-249 | 0,3 | 3,4 | 4 | 3,7 | 66 | 14 | 146 | 99 | 0,3 |
| Cf-250 | 0,6 | 7 | 9 | 8 | 137 | 28 | 302 | 204 | 0,6 |
| Cf-251 | 0,3 | 3,3 | 4 | 3,6 | 66 | 13 | 144 | 98 | 0,3 |
| Cf-252 | 1,1 | 12 | 17 | 14 | 233 | 47 | 513 | 347 | 1,1 |
| Cf-253 | 17 | 202 | 278 | 225 | 3,6E+3 | 728 | 7,9E+3 | 5,3E+3 | 17 |
| Cf-254 | 0,5 | 5 | 7 | 6 | 113 | 23 | 250 | 169 | 0,5 |

| Nuclide | Derived Level Workers [Bq/cm²] W-SM | W-SR | W-LR | W-FF | Derived Level Public [Bq/cm²] P-SM | P-SR | P-LR | P-FF | Overall Min. |
|---|---|---|---|---|---|---|---|---|---|
| Cl-36 | 1,8E+3 | 4,5E+3 | 4,6E+3 | 4,5E+3 | 6,4E+5 | 1,3E+5 | 1,4E+6 | 9,5E+5 | 1,8E+3 |
| Cl-38 | 194 | 193 | 412 | 531 | 2,8E+5 | 2,9E+4 | 6,0E+3 | 1,2E+4 | 193 |
| Cm-240 | 7 | 81 | 110 | 90 | 1,5E+3 | 296 | 3,2E+3 | 2,2E+3 | 7 |
| Cm-241 | 303 | 474 | 1,0E+3 | 1,1E+3 | 1,1E+5 | 1,9E+4 | 1,5E+4 | 2,6E+4 | 303 |
| Cm-242 | 4 | 49 | 67 | 55 | 895 | 182 | 2,0E+3 | 1,3E+3 | 4 |
| Cm-243 | 0,7 | 8 | 10 | 8 | 150 | 31 | 329 | 223 | 0,7 |
| Cm-244 | 0,8 | 9 | 12 | 10 | 172 | 35 | 380 | 257 | 0,8 |
| Cm-245 | 0,5 | 6 | 7 | 6 | 111 | 23 | 244 | 165 | 0,5 |
| Cm-246 | 0,5 | 6 | 7 | 6 | 111 | 23 | 244 | 165 | 0,5 |
| Cm-247 | 0,6 | 6 | 8 | 7 | 119 | 24 | 260 | 177 | 0,6 |
| Cm-248 | 0,1 | 1,6 | 2,1 | 1,7 | 31 | 6 | 68 | 46 | 0,1 |
| Co-55 | 135 | 134 | 305 | 407 | 1,8E+5 | 1,9E+4 | 4,0E+3 | 7,6E+3 | 134 |
| Co-56 | 81 | 80 | 185 | 249 | 9,7E+4 | 1,1E+4 | 2,3E+3 | 4,5E+3 | 80 |
| Co-57 | 2,3E+3 | 2,2E+3 | 4,9E+3 | 6,4E+3 | 2,3E+6 | 2,8E+5 | 6,8E+4 | 1,3E+5 | 2,2E+3 |
| Co-58 | 270 | 268 | 621 | 834 | 3,2E+5 | 3,6E+4 | 7,9E+3 | 1,5E+4 | 268 |
| Co-58m | 1,2E+5 | 1,3E+5 | 1,3E+5 | 1,3E+5 | 3,5E+8 | 7,1E+7 | 3,1E+8 | 3,5E+8 | 1,2E+5 |
| Co-60 | 109 | 108 | 245 | 323 | 1,1E+5 | 1,3E+4 | 3,2E+3 | 6,1E+3 | 108 |
| Cr-51 | 8,2E+3 | 8,2E+3 | 1,8E+4 | 2,4E+4 | 1,0E+7 | 1,1E+6 | 2,4E+5 | 4,7E+5 | 8,2E+3 |
| Cs-129 | 893 | 886 | 2,1E+3 | 2,8E+3 | 1,2E+6 | 1,2E+5 | 2,6E+4 | 4,9E+4 | 886 |
| Cs-131 | 7,7E+3 | 7,7E+3 | 1,7E+4 | 2,2E+4 | 1,0E+7 | 1,1E+6 | 2,4E+5 | 4,6E+5 | 7,7E+3 |
| Cs-132 | 362 | 359 | 836 | 1,1E+3 | 4,7E+5 | 5,0E+4 | 1,0E+4 | 2,0E+4 | 359 |
| Cs-134 | 129 | 128 | 227 | 266 | 1,7E+5 | 2,0E+4 | 4,9E+3 | 9,3E+3 | 128 |
| Cs-134m | 6,4E+3 | 6,4E+3 | 1,1E+4 | 1,2E+4 | 1,2E+7 | 1,2E+6 | 2,6E+5 | 5,0E+5 | 6,4E+3 |
| Cs-135 | 4,0E+3 | 4,6E+3 | 4,6E+3 | 4,6E+3 | 6,7E+6 | 1,4E+6 | 1,5E+7 | 1,0E+7 | 4,0E+3 |
| Cs-136 | 122 | 121 | 274 | 363 | 1,6E+5 | 1,7E+4 | 3,6E+3 | 6,9E+3 | 121 |
| Cs-137 | 284 | 283 | 439 | 487 | 3,8E+5 | 5,0E+4 | 1,3E+4 | 2,5E+4 | 283 |
| Cu-64 | 1,3E+3 | 1,3E+3 | 2,8E+3 | 3,6E+3 | 1,8E+6 | 1,9E+5 | 4,0E+4 | 7,6E+4 | 1,3E+3 |
| Cu-67 | 1,5E+3 | 1,5E+3 | 2,4E+3 | 2,7E+3 | 2,3E+6 | 2,9E+5 | 7,0E+4 | 1,3E+5 | 1,5E+3 |
| Dy-159 | 4,8E+3 | 4,8E+3 | 1,1E+4 | 1,4E+4 | 4,3E+6 | 5,5E+5 | 1,4E+5 | 2,7E+5 | 4,8E+3 |
| Dy-165 | 3,9E+3 | 3,9E+3 | 5,0E+3 | 5,2E+3 | 1,1E+7 | 1,3E+6 | 2,9E+5 | 5,6E+5 | 3,9E+3 |
| Dy-166 | 1,5E+3 | 1,6E+3 | 2,1E+3 | 2,2E+3 | 1,3E+6 | 2,2E+5 | 1,1E+5 | 1,9E+5 | 1,5E+3 |
| Er-169 | 8,7E+3 | 1,4E+4 | 1,4E+4 | 1,4E+4 | 4,7E+6 | 9,5E+5 | 1,0E+7 | 6,9E+6 | 8,7E+3 |
| Er-171 | 647 | 642 | 1,3E+3 | 1,7E+3 | 9,1E+5 | 9,9E+4 | 2,1E+4 | 4,0E+4 | 642 |
| Eu-147 | 322 | 320 | 745 | 1,0E+3 | 3,9E+5 | 4,3E+4 | 9,3E+3 | 1,8E+4 | 320 |
| Eu-148 | 124 | 123 | 289 | 392 | 1,5E+5 | 1,7E+4 | 3,6E+3 | 6,9E+3 | 123 |
| Eu-149 | 3,2E+3 | 3,2E+3 | 7,2E+3 | 9,5E+3 | 3,4E+6 | 4,0E+5 | 9,4E+4 | 1,8E+5 | 3,2E+3 |
| Eu-150 | 331 | 1,5E+3 | 1,7E+3 | 1,6E+3 | 8,7E+4 | 1,7E+4 | 8,9E+4 | 9,2E+4 | 331 |
| Eu-152 | 188 | 226 | 517 | 629 | 8,2E+4 | 1,3E+4 | 6,7E+3 | 1,2E+4 | 188 |
| Eu-152m | 741 | 736 | 1,5E+3 | 1,8E+3 | 1,1E+6 | 1,2E+5 | 2,5E+4 | 4,7E+4 | 736 |
| Eu-154 | 160 | 204 | 456 | 542 | 6,7E+4 | 1,1E+4 | 6,1E+3 | 1,1E+4 | 160 |
| Eu-155 | 1,9E+3 | 3,6E+3 | 7,0E+3 | 7,4E+3 | 6,1E+5 | 1,1E+5 | 1,2E+5 | 2,0E+5 | 1,9E+3 |
| Eu-156 | 200 | 198 | 425 | 545 | 2,4E+5 | 2,7E+4 | 6,2E+3 | 1,2E+4 | 198 |
| F-18 | 260 | 258 | 593 | 797 | 3,5E+5 | 3,7E+4 | 7,6E+3 | 1,5E+4 | 258 |
| Fe-52 | 85 | 85 | 198 | 268 | 1,1E+5 | 1,2E+4 | 2,5E+3 | 4,7E+3 | 85 |
| Fe-55 | 1,9E+4 | 2,7E+4 | 2,7E+4 | 2,7E+4 | 1,2E+7 | 2,5E+6 | 2,7E+7 | 1,8E+7 | 1,9E+4 |
| Fe-59 | 228 | 227 | 509 | 668 | 2,5E+5 | 2,9E+4 | 6,8E+3 | 1,3E+4 | 227 |
| Fe-60 | 57 | 87 | 88 | 87 | 3,3E+4 | 6,8E+3 | 7,3E+4 | 5,0E+4 | 57 |
| Ga-67 | 1,7E+3 | 1,7E+3 | 3,5E+3 | 4,4E+3 | 2,2E+6 | 2,5E+5 | 5,5E+4 | 1,0E+5 | 1,7E+3 |
| Ga-68 | 260 | 258 | 564 | 734 | 3,6E+5 | 3,9E+4 | 7,9E+3 | 1,5E+4 | 258 |
| Ga-72 | 107 | 106 | 243 | 326 | 1,4E+5 | 1,5E+4 | 3,1E+3 | 6,0E+3 | 106 |
| Gd-146 | 97 | 96 | 220 | 292 | 1,1E+5 | 1,3E+4 | 2,9E+3 | 5,5E+3 | 96 |
| Gd-148 | 1,8 | 20 | 27 | 22 | 423 | 86 | 933 | 631 | 1,8 |
| Gd-153 | 2,1E+3 | 2,1E+3 | 4,6E+3 | 5,7E+3 | 1,3E+6 | 1,8E+5 | 6,3E+4 | 1,2E+5 | 2,1E+3 |
| Gd-159 | 3,2E+3 | 3,2E+3 | 4,4E+3 | 4,8E+3 | 5,7E+6 | 7,1E+5 | 1,8E+5 | 3,5E+5 | 3,2E+3 |
| Ge-68 | 266 | 265 | 608 | 784 | 1,7E+5 | 2,5E+4 | 7,9E+3 | 1,5E+4 | 265 |
| Ge-71 | 5,8E+5 | 7,8E+5 | 7,9E+5 | 7,9E+5 | 4,2E+8 | 8,6E+7 | 9,3E+8 | 6,3E+8 | 5,8E+5 |
| Ge-77 | 235 | 233 | 496 | 637 | 3,2E+5 | 3,5E+4 | 7,3E+3 | 1,4E+4 | 233 |
| Hf-172 | 699 | 1,6E+3 | 2,8E+3 | 2,9E+3 | 2,2E+5 | 4,1E+4 | 5,7E+4 | 9,1E+4 | 699 |
| Hf-175 | 690 | 686 | 1,5E+3 | 2,0E+3 | 7,6E+5 | 8,9E+4 | 2,1E+4 | 3,9E+4 | 686 |
| Hf-181 | 447 | 445 | 952 | 1,2E+3 | 3,8E+5 | 5,0E+4 | 1,4E+4 | 2,6E+4 | 445 |

| Nuclide | Derived Level Workers [Bq/cm²] W-SM | W-SR | W-LR | W-FF | Derived Level Public [Bq/cm²] P-SM | P-SR | P-LR | P-FF | Overall Min. |
|---|---|---|---|---|---|---|---|---|---|
| Hf-182 | 142 | 651 | 939 | 911 | 3,5E+4 | 7,0E+3 | 2,3E+4 | 2,9E+4 | 142 |
| Hg-194 | 112 | 112 | 149 | 158 | 2,1E+5 | 2,8E+4 | 7,5E+3 | 1,4E+4 | 112 |
| Hg-195m | 817 | 811 | 1,8E+3 | 2,3E+3 | 1,0E+6 | 1,1E+5 | 2,5E+4 | 4,8E+4 | 811 |
| Hg-197 | 3,4E+3 | 3,4E+3 | 6,5E+3 | 7,9E+3 | 4,0E+6 | 4,9E+5 | 1,2E+5 | 2,3E+5 | 3,4E+3 |
| Hg-197m | 1,4E+3 | 1,4E+3 | 2,6E+3 | 3,1E+3 | 1,7E+6 | 2,1E+5 | 5,2E+4 | 1,0E+5 | 1,4E+3 |
| Hg-203 | 943 | 941 | 1,8E+3 | 2,1E+3 | 8,7E+5 | 1,2E+5 | 3,4E+4 | 6,4E+4 | 941 |
| Ho-166 | 2,2E+3 | 2,3E+3 | 2,6E+3 | 2,7E+3 | 4,6E+6 | 7,0E+5 | 2,7E+5 | 5,1E+5 | 2,2E+3 |
| Ho-166m | 96 | 153 | 350 | 393 | 3,3E+4 | 5,9E+3 | 4,6E+3 | 8,0E+3 | 96 |
| I-123 | 1,5E+3 | 1,5E+3 | 3,3E+3 | 4,4E+3 | 2,0E+6 | 2,1E+5 | 4,4E+4 | 8,4E+4 | 1,5E+3 |
| I-124 | 194 | 193 | 342 | 401 | 2,6E+5 | 3,1E+4 | 7,4E+3 | 1,4E+4 | 193 |
| I-125 | 517 | 577 | 626 | 630 | 7,9E+5 | 1,4E+5 | 1,2E+5 | 2,1E+5 | 517 |
| I-126 | 207 | 210 | 270 | 283 | 2,9E+5 | 4,3E+4 | 1,6E+4 | 3,0E+4 | 207 |
| I-129 | 77 | 89 | 90 | 89 | 1,3E+5 | 2,6E+4 | 1,3E+5 | 1,3E+5 | 77 |
| I-131 | 266 | 269 | 348 | 365 | 3,7E+5 | 5,5E+4 | 2,0E+4 | 3,7E+4 | 266 |
| I-132 | 116 | 116 | 268 | 362 | 1,5E+5 | 1,6E+4 | 3,4E+3 | 6,5E+3 | 116 |
| I-133 | 329 | 327 | 619 | 747 | 4,5E+5 | 5,2E+4 | 1,2E+4 | 2,2E+4 | 327 |
| I-134 | 103 | 103 | 238 | 322 | 1,4E+5 | 1,5E+4 | 3,0E+3 | 5,7E+3 | 103 |
| I-135 | 147 | 146 | 333 | 446 | 2,0E+5 | 2,1E+4 | 4,3E+3 | 8,3E+3 | 146 |
| In-111 | 661 | 656 | 1,5E+3 | 2,0E+3 | 8,5E+5 | 9,2E+4 | 1,9E+4 | 3,7E+4 | 656 |
| In-113m | 953 | 945 | 2,1E+3 | 2,7E+3 | 1,3E+6 | 1,4E+5 | 2,9E+4 | 5,6E+4 | 945 |
| In-114m | 877 | 1,0E+3 | 1,5E+3 | 1,6E+3 | 5,9E+5 | 1,0E+5 | 5,6E+4 | 1,0E+5 | 877 |
| In-115m | 1,4E+3 | 1,4E+3 | 2,9E+3 | 3,7E+3 | 2,0E+6 | 2,2E+5 | 4,5E+4 | 8,7E+4 | 1,4E+3 |
| Ir-189 | 2,9E+3 | 2,9E+3 | 5,7E+3 | 6,9E+3 | 3,0E+6 | 3,8E+5 | 9,8E+4 | 1,9E+5 | 2,9E+3 |
| Ir-190 | 186 | 184 | 423 | 564 | 2,3E+5 | 2,5E+4 | 5,4E+3 | 1,0E+4 | 184 |
| Ir-192 | 305 | 303 | 663 | 851 | 2,9E+5 | 3,6E+4 | 9,3E+3 | 1,8E+4 | 303 |
| Ir-194 | 1,4E+3 | 1,4E+3 | 2,0E+3 | 2,2E+3 | 2,7E+6 | 3,4E+5 | 8,5E+4 | 1,6E+5 | 1,4E+3 |
| K-40 | 761 | 761 | 999 | 1,1E+3 | 1,2E+6 | 1,7E+5 | 5,3E+4 | 1,0E+5 | 761 |
| K-42 | 1,0E+3 | 1,0E+3 | 2,3E+3 | 3,0E+3 | 1,4E+6 | 1,5E+5 | 3,1E+4 | 5,9E+4 | 1,0E+3 |
| K-43 | 266 | 264 | 600 | 800 | 3,6E+5 | 3,8E+4 | 7,8E+3 | 1,5E+4 | 264 |
| La-137 | 3,7E+3 | 7,2E+3 | 1,6E+4 | 1,7E+4 | 1,2E+6 | 2,1E+5 | 2,1E+5 | 3,6E+5 | 3,7E+3 |
| La-140 | 123 | 122 | 280 | 375 | 1,6E+5 | 1,7E+4 | 3,6E+3 | 6,9E+3 | 122 |
| Lu-172 | 144 | 143 | 324 | 431 | 1,8E+5 | 2,0E+4 | 4,3E+3 | 8,2E+3 | 143 |
| Lu-173 | 1,8E+3 | 1,8E+3 | 4,0E+3 | 5,0E+3 | 1,2E+6 | 1,7E+5 | 5,7E+4 | 1,1E+5 | 1,8E+3 |
| Lu-174 | 1,6E+3 | 1,9E+3 | 3,8E+3 | 4,5E+3 | 8,0E+5 | 1,3E+5 | 6,1E+4 | 1,1E+5 | 1,6E+3 |
| Lu-174m | 2,3E+3 | 3,0E+3 | 5,4E+3 | 5,9E+3 | 1,0E+6 | 1,8E+5 | 1,2E+5 | 2,1E+5 | 2,3E+3 |
| Lu-177 | 3,9E+3 | 4,2E+3 | 6,1E+3 | 6,5E+3 | 3,0E+6 | 4,8E+5 | 2,2E+5 | 4,1E+5 | 3,9E+3 |
| Mg-28 | 90 | 90 | 203 | 269 | 1,2E+5 | 1,3E+4 | 2,7E+3 | 5,1E+3 | 90 |
| Mn-52 | 80 | 79 | 185 | 251 | 1,0E+5 | 1,1E+4 | 2,3E+3 | 4,4E+3 | 79 |
| Mn-53 | 1,8E+5 | 3,1E+5 | 3,2E+5 | 3,1E+5 | 8,6E+7 | 1,8E+7 | 1,9E+8 | 1,3E+8 | 1,8E+5 |
| Mn-54 | 317 | 315 | 730 | 980 | 3,7E+5 | 4,2E+4 | 9,2E+3 | 1,8E+4 | 315 |
| Mn-56 | 166 | 164 | 370 | 492 | 2,2E+5 | 2,4E+4 | 4,9E+3 | 9,4E+3 | 164 |
| Mo-93 | 2,4E+3 | 3,2E+3 | 3,5E+3 | 3,5E+3 | 1,9E+6 | 3,6E+5 | 4,9E+5 | 7,9E+5 | 2,4E+3 |
| Mo-99 | 846 | 841 | 1,6E+3 | 1,9E+3 | 1,1E+6 | 1,2E+5 | 2,9E+4 | 5,6E+4 | 841 |
| N-13 | 260 | 258 | 619 | 853 | 3,4E+5 | 3,6E+4 | 7,4E+3 | 1,4E+4 | 258 |
| Na-22 | 121 | 120 | 269 | 355 | 1,6E+5 | 1,7E+4 | 3,6E+3 | 6,9E+3 | 120 |
| Na-24 | 78 | 77 | 180 | 245 | 1,0E+5 | 1,1E+4 | 2,2E+3 | 4,3E+3 | 77 |
| Nb-93m | 2,4E+4 | 4,5E+4 | 6,0E+4 | 6,1E+4 | 8,6E+6 | 1,7E+6 | 2,7E+6 | 4,1E+6 | 2,4E+4 |
| Nb-94 | 165 | 164 | 374 | 490 | 1,5E+5 | 1,9E+4 | 4,9E+3 | 9,2E+3 | 164 |
| Nb-95 | 344 | 341 | 791 | 1,1E+3 | 4,0E+5 | 4,5E+4 | 1,0E+4 | 1,9E+4 | 341 |
| Nb-97 | 377 | 374 | 820 | 1,1E+3 | 5,2E+5 | 5,5E+4 | 1,1E+4 | 2,2E+4 | 374 |
| Nd-147 | 1,3E+3 | 1,3E+3 | 2,3E+3 | 2,6E+3 | 1,2E+6 | 1,6E+5 | 5,2E+4 | 9,8E+4 | 1,3E+3 |
| Nd-149 | 579 | 575 | 1,1E+3 | 1,4E+3 | 7,8E+5 | 8,8E+4 | 2,0E+4 | 3,7E+4 | 575 |
| Ni-59 | 7,8E+4 | 1,5E+5 | 1,5E+5 | 1,5E+5 | 3,6E+7 | 7,3E+6 | 7,9E+7 | 5,3E+7 | 7,8E+4 |
| Ni-63 | 2,7E+4 | 5,9E+4 | 6,1E+4 | 5,9E+4 | 9,7E+6 | 2,0E+6 | 2,1E+7 | 1,4E+7 | 2,7E+4 |
| Ni-65 | 481 | 477 | 1,0E+3 | 1,3E+3 | 6,8E+5 | 7,3E+4 | 1,5E+4 | 2,9E+4 | 477 |
| Np-235 | 3,3E+4 | 8,1E+4 | 1,1E+5 | 1,1E+5 | 1,1E+7 | 2,1E+6 | 4,0E+6 | 5,9E+6 | 3,3E+4 |
| Np-236 | 7 | 71 | 93 | 78 | 1,5E+3 | 295 | 3,1E+3 | 2,1E+3 | 7 |
| Np-237 | 0,9 | 11 | 14 | 12 | 202 | 41 | 440 | 300 | 0,9 |
| Np-239 | 1,4E+3 | 1,4E+3 | 2,5E+3 | 3,0E+3 | 1,6E+6 | 1,9E+5 | 4,9E+4 | 9,3E+4 | 1,4E+3 |
| Os-185 | 372 | 369 | 859 | 1,2E+3 | 4,4E+5 | 4,9E+4 | 1,1E+4 | 2,1E+4 | 369 |

| | Derived Level Workers [Bq/cm²] | | | | Derived Level Public [Bq/cm²] | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Nuclide | W-SM | W-SR | W-LR | W-FF | P-SM | P-SR | P-LR | P-FF | Overall Min. |
| Os-191 | 2,9E+3 | 3,0E+3 | 5,6E+3 | 6,4E+3 | 1,8E+6 | 2,8E+5 | 1,1E+5 | 2,0E+5 | 2,9E+3 |
| Os-191m | 1,8E+4 | 1,9E+4 | 2,3E+4 | 2,4E+4 | 2,3E+7 | 3,7E+6 | 1,6E+6 | 3,0E+6 | 1,8E+4 |
| Os-193 | 2,2E+3 | 2,2E+3 | 3,2E+3 | 3,5E+3 | 3,4E+6 | 4,3E+5 | 1,1E+5 | 2,2E+5 | 2,2E+3 |
| Os-194 | 605 | 1,3E+3 | 1,8E+3 | 1,8E+3 | 2,0E+5 | 4,0E+4 | 7,2E+4 | 1,1E+5 | 605 |
| P-32 | 1,7E+3 | 2,2E+3 | 2,2E+3 | 2,2E+3 | 1,4E+6 | 2,8E+5 | 3,0E+6 | 2,0E+6 | 1,7E+3 |
| P-33 | 8,7E+3 | 2,0E+4 | 2,0E+4 | 2,0E+4 | 3,1E+6 | 6,3E+5 | 6,8E+6 | 4,6E+6 | 8,7E+3 |
| Pa-230 | 34 | 269 | 386 | 352 | 7,5E+3 | 1,5E+3 | 7,0E+3 | 7,6E+3 | 34 |
| Pa-231 | 0,2 | 1,7 | 2,3 | 1,9 | 33 | 7 | 73 | 50 | 0,2 |
| Pa-233 | 1,1E+3 | 1,1E+3 | 2,2E+3 | 2,6E+3 | 7,7E+5 | 1,1E+5 | 3,7E+4 | 6,9E+4 | 1,1E+3 |
| Pb-201 | 309 | 307 | 717 | 971 | 4,1E+5 | 4,3E+4 | 8,9E+3 | 1,7E+4 | 307 |
| Pb-202 | 705 | 1,1E+3 | 1,1E+3 | 1,1E+3 | 4,2E+5 | 8,6E+4 | 9,3E+5 | 6,3E+5 | 705 |
| Pb-203 | 877 | 870 | 2,0E+3 | 2,6E+3 | 1,2E+6 | 1,2E+5 | 2,6E+4 | 5,0E+4 | 870 |
| Pb-205 | 2,2E+4 | 3,3E+4 | 3,4E+4 | 3,4E+4 | 1,4E+7 | 2,9E+6 | 3,1E+7 | 2,1E+7 | 2,2E+4 |
| Pb-210 | 3,4 | 9 | 10 | 9 | 1,1E+3 | 220 | 2,4E+3 | 1,6E+3 | 3,4 |
| Pb-212 | 140 | 167 | 337 | 391 | 7,0E+4 | 1,1E+4 | 5,6E+3 | 1,0E+4 | 140 |
| Pd-103 | 9,5E+3 | 9,5E+3 | 1,8E+4 | 2,2E+4 | 6,8E+6 | 9,9E+5 | 3,3E+5 | 6,2E+5 | 9,5E+3 |
| Pd-107 | 1,3E+5 | 2,5E+5 | 2,5E+5 | 2,5E+5 | 5,5E+7 | 1,1E+7 | 1,2E+8 | 8,2E+7 | 1,3E+5 |
| Pd-109 | 3,8E+3 | 3,9E+3 | 4,4E+3 | 4,5E+3 | 8,6E+6 | 1,3E+6 | 5,0E+5 | 9,4E+5 | 3,8E+3 |
| Pm-143 | 817 | 812 | 1,9E+3 | 2,5E+3 | 8,1E+5 | 9,7E+4 | 2,4E+4 | 4,5E+4 | 812 |
| Pm-144 | 167 | 167 | 391 | 523 | 1,6E+5 | 2,0E+4 | 4,8E+3 | 9,2E+3 | 167 |
| Pm-145 | 3,3E+3 | 5,6E+3 | 1,2E+4 | 1,3E+4 | 1,1E+6 | 2,0E+5 | 1,8E+5 | 3,1E+5 | 3,3E+3 |
| Pm-147 | 3,6E+3 | 1,6E+4 | 1,8E+4 | 1,7E+4 | 9,3E+5 | 1,9E+5 | 2,1E+6 | 1,4E+6 | 3,6E+3 |
| Pm-148m | 126 | 126 | 281 | 370 | 1,5E+5 | 1,7E+4 | 3,8E+3 | 7,2E+3 | 126 |
| Pm-149 | 3,4E+3 | 3,7E+3 | 4,1E+3 | 4,1E+3 | 5,7E+6 | 9,8E+5 | 6,4E+5 | 1,1E+6 | 3,4E+3 |
| Pm-151 | 654 | 650 | 1,2E+3 | 1,5E+3 | 9,5E+5 | 1,1E+5 | 2,3E+4 | 4,4E+4 | 650 |
| Po-210 | 6 | 28 | 31 | 29 | 1,4E+3 | 287 | 3,1E+3 | 2,1E+3 | 6 |
| Pr-142 | 2,6E+3 | 2,6E+3 | 3,7E+3 | 4,0E+3 | 3,8E+6 | 5,1E+5 | 1,4E+5 | 2,7E+5 | 2,6E+3 |
| Pr-143 | 2,8E+3 | 3,9E+3 | 3,9E+3 | 3,9E+3 | 2,1E+6 | 4,3E+5 | 4,7E+6 | 3,2E+6 | 2,8E+3 |
| Pt-188 | 161 | 160 | 363 | 484 | 2,1E+5 | 2,3E+4 | 4,7E+3 | 9,1E+3 | 160 |
| Pt-191 | 887 | 879 | 2,0E+3 | 2,6E+3 | 1,2E+6 | 1,3E+5 | 2,7E+4 | 5,1E+4 | 879 |
| Pt-193 | 2,0E+5 | 2,4E+5 | 2,5E+5 | 2,4E+5 | 2,2E+8 | 4,5E+7 | 4,9E+8 | 3,3E+8 | 2,0E+5 |
| Pt-193m | 7,2E+3 | 7,2E+3 | 8,6E+3 | 8,9E+3 | 1,8E+7 | 2,5E+6 | 7,3E+5 | 1,4E+6 | 7,2E+3 |
| Pt-195m | 1,9E+3 | 1,9E+3 | 2,6E+3 | 2,8E+3 | 4,3E+6 | 5,0E+5 | 1,1E+5 | 2,1E+5 | 1,9E+3 |
| Pt-197 | 4,5E+3 | 4,5E+3 | 5,8E+3 | 6,1E+3 | 1,2E+7 | 1,5E+6 | 3,5E+5 | 6,6E+5 | 4,5E+3 |
| Pt-197m | 1,9E+3 | 1,9E+3 | 3,1E+3 | 3,6E+3 | 3,2E+6 | 3,6E+5 | 7,6E+4 | 1,5E+5 | 1,9E+3 |
| Pu-236 | 1,1 | 12 | 17 | 14 | 233 | 47 | 513 | 347 | 1,1 |
| Pu-237 | 5,7E+3 | 5,7E+3 | 1,2E+4 | 1,6E+4 | 5,0E+6 | 6,5E+5 | 1,8E+5 | 3,3E+5 | 5,7E+3 |
| Pu-238 | 0,5 | 5 | 7 | 6 | 101 | 21 | 223 | 151 | 0,5 |
| Pu-239 | 0,4 | 5 | 6 | 5 | 93 | 19 | 205 | 139 | 0,4 |
| Pu-240 | 0,4 | 5 | 6 | 5 | 93 | 19 | 205 | 139 | 0,4 |
| Pu-241 | 23 | 260 | 345 | 286 | 5,2E+3 | 1,1E+3 | 1,1E+4 | 7,7E+3 | 23 |
| Pu-242 | 0,5 | 5 | 7 | 6 | 97 | 20 | 214 | 145 | 0,5 |
| Pu-244 | 0,5 | 5 | 7 | 6 | 99 | 20 | 216 | 147 | 0,5 |
| Ra-223 | 2,8 | 26 | 33 | 28 | 628 | 128 | 1,3E+3 | 920 | 2,8 |
| Ra-224 | 6 | 48 | 59 | 53 | 1,5E+3 | 309 | 2,3E+3 | 2,0E+3 | 6 |
| Ra-225 | 2,9 | 26 | 32 | 28 | 645 | 131 | 1,4E+3 | 950 | 2,9 |
| Ra-226 | 2,2 | 7 | 7 | 7 | 619 | 126 | 1,1E+3 | 839 | 2,2 |
| Ra-228 | 5 | 13 | 13 | 13 | 1,8E+3 | 357 | 2,7E+3 | 2,3E+3 | 5 |
| Rb-81 | 416 | 412 | 903 | 1,2E+3 | 5,8E+5 | 6,1E+4 | 1,3E+4 | 2,4E+4 | 412 |
| Rb-83 | 480 | 477 | 1,0E+3 | 1,3E+3 | 6,3E+5 | 7,0E+4 | 1,5E+4 | 2,9E+4 | 477 |
| Rb-84 | 272 | 270 | 565 | 717 | 3,6E+5 | 4,0E+4 | 8,6E+3 | 1,6E+4 | 270 |
| Rb-86 | 1,2E+3 | 1,2E+3 | 1,6E+3 | 1,7E+3 | 2,2E+6 | 2,9E+5 | 8,4E+4 | 1,6E+5 | 1,2E+3 |
| Rb-87 | 3,0E+3 | 3,2E+3 | 3,2E+3 | 3,2E+3 | 9,3E+6 | 1,9E+6 | 2,1E+7 | 1,4E+7 | 3,0E+3 |
| Re-184 | 292 | 290 | 656 | 867 | 3,4E+5 | 3,9E+4 | 8,7E+3 | 1,7E+4 | 290 |
| Re-184m | 235 | 234 | 500 | 632 | 2,1E+5 | 2,8E+4 | 7,3E+3 | 1,4E+4 | 234 |
| Re-186 | 2,6E+3 | 2,9E+3 | 3,3E+3 | 3,4E+3 | 3,4E+6 | 5,9E+5 | 3,8E+5 | 6,7E+5 | 2,6E+3 |
| Re-187 | 1,2E+6 | 1,9E+6 | 1,9E+6 | 1,9E+6 | 7,4E+8 | 1,5E+8 | 1,6E+9 | 1,1E+9 | 1,2E+6 |
| Re-188 | 1,8E+3 | 1,8E+3 | 2,2E+3 | 2,4E+3 | 3,6E+6 | 4,8E+5 | 1,4E+5 | 2,6E+5 | 1,8E+3 |
| Re-189 | 1,7E+3 | 1,7E+3 | 2,2E+3 | 2,4E+3 | 3,6E+6 | 4,5E+5 | 1,1E+5 | 2,2E+5 | 1,7E+3 |
| Rh-99 | 424 | 421 | 952 | 1,3E+3 | 5,3E+5 | 5,8E+4 | 1,3E+4 | 2,4E+4 | 421 |

| Nuclide | Derived Level Workers [Bq/cm²] W-SM | W-SR | W-LR | W-FF | Derived Level Public [Bq/cm²] P-SM | P-SR | P-LR | P-FF | Overall Min. |
|---|---|---|---|---|---|---|---|---|---|
| Rh-101 | 936 | 932 | 2,0E+3 | 2,5E+3 | 8,1E+5 | 1,1E+5 | 2,9E+4 | 5,5E+4 | 932 |
| Rh-102 | 123 | 122 | 280 | 372 | 1,3E+5 | 1,6E+4 | 3,6E+3 | 6,9E+3 | 122 |
| Rh-102m | 502 | 500 | 1,1E+3 | 1,4E+3 | 4,4E+5 | 5,7E+4 | 1,6E+4 | 2,9E+4 | 500 |
| Rh-103m | 1,1E+5 | 1,1E+5 | 2,4E+5 | 3,1E+5 | 1,4E+8 | 1,6E+7 | 3,3E+6 | 6,4E+6 | 1,1E+5 |
| Rh-105 | 2,6E+3 | 2,6E+3 | 4,6E+3 | 5,4E+3 | 3,5E+6 | 4,2E+5 | 9,9E+4 | 1,9E+5 | 2,6E+3 |
| Rn-222 | 155 | 155 | 349 | 451 | 1,2E+5 | 1,6E+4 | 4,6E+3 | 8,8E+3 | 155 |
| Ru-97 | 1,1E+3 | 1,1E+3 | 2,6E+3 | 3,5E+3 | 1,4E+6 | 1,5E+5 | 3,2E+4 | 6,2E+4 | 1,1E+3 |
| Ru-103 | 516 | 513 | 1,2E+3 | 1,5E+3 | 5,2E+5 | 6,3E+4 | 1,5E+4 | 2,9E+4 | 513 |
| Ru-105 | 290 | 288 | 634 | 829 | 3,9E+5 | 4,2E+4 | 8,8E+3 | 1,7E+4 | 288 |
| Ru-106 | 352 | 560 | 764 | 781 | 1,5E+5 | 2,8E+4 | 3,3E+4 | 5,5E+4 | 352 |
| S-35 | 6,5E+3 | 1,1E+4 | 1,1E+4 | 1,1E+4 | 3,3E+6 | 6,8E+5 | 7,3E+6 | 5,0E+6 | 6,5E+3 |
| Sb-122 | 501 | 498 | 970 | 1,2E+3 | 6,7E+5 | 7,6E+4 | 1,7E+4 | 3,3E+4 | 498 |
| Sb-124 | 145 | 144 | 322 | 421 | 1,6E+5 | 1,9E+4 | 4,4E+3 | 8,3E+3 | 144 |
| Sb-125 | 536 | 535 | 1,1E+3 | 1,4E+3 | 4,0E+5 | 5,5E+4 | 1,7E+4 | 3,2E+4 | 535 |
| Sb-126 | 94 | 93 | 211 | 280 | 1,2E+5 | 1,3E+4 | 2,8E+3 | 5,3E+3 | 93 |
| Sc-44 | 124 | 123 | 277 | 369 | 1,7E+5 | 1,8E+4 | 3,7E+3 | 7,0E+3 | 123 |
| Sc-46 | 135 | 134 | 310 | 413 | 1,4E+5 | 1,7E+4 | 3,9E+3 | 7,5E+3 | 134 |
| Sc-47 | 2,0E+3 | 2,0E+3 | 3,6E+3 | 4,2E+3 | 2,3E+6 | 2,9E+5 | 7,5E+4 | 1,4E+5 | 2,0E+3 |
| Sc-48 | 83 | 82 | 190 | 257 | 1,1E+5 | 1,1E+4 | 2,4E+3 | 4,6E+3 | 82 |
| Se-75 | 603 | 599 | 1,2E+3 | 1,4E+3 | 7,8E+5 | 9,0E+4 | 2,0E+4 | 3,9E+4 | 599 |
| Se-79 | 2,5E+3 | 2,9E+3 | 2,9E+3 | 2,9E+3 | 4,2E+6 | 8,6E+5 | 9,3E+6 | 6,3E+6 | 2,5E+3 |
| Si-31 | 5,3E+3 | 5,4E+3 | 5,4E+3 | 5,4E+3 | 5,5E+7 | 1,0E+7 | 8,7E+6 | 1,5E+7 | 5,3E+3 |
| Si-32 | 1,1E+3 | 4,5E+3 | 4,9E+3 | 4,6E+3 | 2,7E+5 | 5,6E+4 | 6,0E+5 | 4,1E+5 | 1,1E+3 |
| Sm-145 | 3,1E+3 | 3,1E+3 | 6,8E+3 | 8,5E+3 | 1,7E+6 | 2,6E+5 | 9,3E+4 | 1,7E+5 | 3,1E+3 |
| Sm-147 | 2,2 | 25 | 33 | 27 | 485 | 99 | 1,1E+3 | 723 | 2,2 |
| Sm-151 | 5,1E+3 | 4,0E+4 | 4,8E+4 | 4,3E+4 | 1,2E+6 | 2,4E+5 | 2,6E+6 | 1,7E+6 | 5,1E+3 |
| Sm-153 | 2,4E+3 | 2,4E+3 | 3,6E+3 | 3,9E+3 | 3,1E+6 | 4,1E+5 | 1,2E+5 | 2,2E+5 | 2,4E+3 |
| Sn-113 | 849 | 845 | 1,9E+3 | 2,4E+3 | 7,0E+5 | 9,3E+4 | 2,6E+4 | 4,9E+4 | 845 |
| Sn-117m | 1,2E+3 | 1,2E+3 | 1,9E+3 | 2,1E+3 | 1,0E+6 | 1,5E+5 | 4,9E+4 | 9,2E+4 | 1,2E+3 |
| Sn-119m | 4,3E+3 | 6,7E+3 | 8,8E+3 | 9,0E+3 | 1,9E+6 | 3,6E+5 | 4,4E+5 | 7,2E+5 | 4,3E+3 |
| Sn-121m | 3,0E+3 | 7,5E+3 | 8,3E+3 | 8,2E+3 | 9,8E+5 | 1,9E+5 | 7,4E+5 | 8,8E+5 | 3,0E+3 |
| Sn-123 | 1,1E+3 | 1,8E+3 | 1,8E+3 | 1,8E+3 | 5,7E+5 | 1,1E+5 | 6,0E+5 | 6,2E+5 | 1,1E+3 |
| Sn-125 | 617 | 616 | 1,0E+3 | 1,1E+3 | 6,7E+5 | 9,1E+4 | 2,6E+4 | 5,0E+4 | 616 |
| Sn-126 | 456 | 1,0E+3 | 1,2E+3 | 1,2E+3 | 1,6E+5 | 3,2E+4 | 9,7E+4 | 1,2E+5 | 456 |
| Sr-82 | 1,4E+3 | 1,6E+3 | 1,6E+3 | 1,6E+3 | 2,2E+6 | 4,5E+5 | 4,9E+6 | 3,3E+6 | 1,4E+3 |
| Sr-85 | 503 | 499 | 1,1E+3 | 1,5E+3 | 6,4E+5 | 7,0E+4 | 1,5E+4 | 2,8E+4 | 499 |
| Sr-85m | 1,3E+3 | 1,3E+3 | 3,0E+3 | 4,1E+3 | 1,7E+6 | 1,8E+5 | 3,7E+4 | 7,1E+4 | 1,3E+3 |
| Sr-87m | 818 | 811 | 1,9E+3 | 2,5E+3 | 1,1E+6 | 1,2E+5 | 2,4E+4 | 4,6E+4 | 811 |
| Sr-89 | 2,0E+3 | 2,2E+3 | 2,2E+3 | 2,2E+3 | 4,6E+6 | 9,4E+5 | 8,7E+6 | 6,5E+6 | 2,0E+3 |
| Sr-90 | 216 | 288 | 290 | 289 | 1,8E+5 | 3,7E+4 | 4,0E+5 | 2,7E+5 | 216 |
| Sr-91 | 365 | 362 | 770 | 989 | 5,1E+5 | 5,5E+4 | 1,1E+4 | 2,2E+4 | 362 |
| Sr-92 | 177 | 175 | 403 | 540 | 2,3E+5 | 2,5E+4 | 5,2E+3 | 9,9E+3 | 175 |
| H-3 | 1,6E+5 | 2,3E+5 | 2,3E+5 | 2,3E+5 | 1,0E+8 | 2,1E+7 | 2,3E+8 | 1,5E+8 | 1,6E+5 |
| Ta-178-l | 263 | 261 | 617 | 843 | 3,4E+5 | 3,7E+4 | 7,5E+3 | 1,4E+4 | 261 |
| Ta-179 | 7,7E+3 | 7,6E+3 | 1,7E+4 | 2,2E+4 | 7,1E+6 | 8,9E+5 | 2,3E+5 | 4,4E+5 | 7,6E+3 |
| Ta-182 | 207 | 206 | 463 | 602 | 2,0E+5 | 2,4E+4 | 6,2E+3 | 1,2E+4 | 206 |
| Tb-157 | 1,4E+4 | 4,0E+4 | 7,2E+4 | 7,4E+4 | 3,7E+6 | 7,1E+5 | 1,3E+6 | 1,9E+6 | 1,4E+4 |
| Tb-158 | 215 | 305 | 668 | 766 | 8,3E+4 | 1,4E+4 | 9,4E+3 | 1,7E+4 | 215 |
| Tb-160 | 239 | 238 | 525 | 676 | 2,2E+5 | 2,8E+4 | 7,3E+3 | 1,4E+4 | 238 |
| Tc-95m | 373 | 370 | 854 | 1,1E+3 | 4,6E+5 | 5,0E+4 | 1,1E+4 | 2,1E+4 | 370 |
| Tc-96 | 106 | 105 | 243 | 327 | 1,4E+5 | 1,5E+4 | 3,1E+3 | 5,9E+3 | 105 |
| Tc-96m | 107 | 106 | 251 | 342 | 1,4E+5 | 1,5E+4 | 3,1E+3 | 5,9E+3 | 106 |
| Tc-97 | 1,5E+4 | 1,5E+4 | 3,0E+4 | 3,6E+4 | 1,1E+7 | 1,5E+6 | 4,8E+5 | 9,1E+5 | 1,5E+4 |
| Tc-97m | 3,4E+3 | 6,4E+3 | 7,9E+3 | 7,9E+3 | 1,4E+6 | 2,7E+5 | 5,0E+5 | 7,4E+5 | 3,4E+3 |
| Tc-98 | 177 | 176 | 381 | 487 | 1,7E+5 | 2,2E+4 | 5,4E+3 | 1,0E+4 | 176 |
| Tc-99 | 3,6E+3 | 1,0E+4 | 1,1E+4 | 1,1E+4 | 1,2E+6 | 2,4E+5 | 2,6E+6 | 1,7E+6 | 3,6E+3 |
| Tc-99m | 2,3E+3 | 2,2E+3 | 5,2E+3 | 7,1E+3 | 2,9E+6 | 3,1E+5 | 6,5E+4 | 1,2E+5 | 2,2E+3 |
| Te-121 | 445 | 441 | 1,0E+3 | 1,4E+3 | 5,6E+5 | 6,1E+4 | 1,3E+4 | 2,5E+4 | 441 |
| Te-121m | 946 | 959 | 1,8E+3 | 2,0E+3 | 6,6E+5 | 9,8E+4 | 3,5E+4 | 6,6E+4 | 946 |
| Te-123m | 1,3E+3 | 1,4E+3 | 2,5E+3 | 2,9E+3 | 7,9E+5 | 1,2E+5 | 5,2E+4 | 9,6E+4 | 1,3E+3 |

| Nuclide | Derived Level Workers [Bq/cm²] | | | | Derived Level Public [Bq/cm²] | | | | Overall Min. |
|---|---|---|---|---|---|---|---|---|---|
| | W-SM | W-SR | W-LR | W-FF | P-SM | P-SR | P-LR | P-FF | |
| Te-125m | 2,3E+3 | 3,1E+3 | 4,8E+3 | 5,1E+3 | 1,1E+6 | 2,0E+5 | 1,5E+5 | 2,6E+5 | 2,3E+3 |
| Te-127 | 7,7E+3 | 7,8E+3 | 8,5E+3 | 8,6E+3 | 2,4E+7 | 3,7E+6 | 1,5E+6 | 2,9E+6 | 7,7E+3 |
| Te-127m | 1,4E+3 | 2,7E+3 | 3,1E+3 | 3,1E+3 | 6,0E+5 | 1,2E+5 | 2,9E+5 | 4,0E+5 | 1,4E+3 |
| Te-129 | 2,5E+3 | 2,5E+3 | 3,7E+3 | 4,0E+3 | 5,6E+6 | 6,1E+5 | 1,3E+5 | 2,5E+5 | 2,5E+3 |
| Te-129m | 1,1E+3 | 1,4E+3 | 1,9E+3 | 2,0E+3 | 6,1E+5 | 1,1E+5 | 9,0E+4 | 1,6E+5 | 1,1E+3 |
| Te-131m | 106 | 106 | 186 | 217 | 1,4E+5 | 1,7E+4 | 4,1E+3 | 7,8E+3 | 106 |
| Te-132 | 103 | 103 | 232 | 308 | 1,3E+5 | 1,4E+4 | 3,1E+3 | 5,9E+3 | 103 |
| Th-227 | 2,1 | 25 | 35 | 28 | 465 | 95 | 1,0E+3 | 690 | 2,1 |
| Th-228 | 0,5 | 5 | 7 | 6 | 108 | 22 | 229 | 159 | 0,5 |
| Th-229 | 0,3 | 2,9 | 3,8 | 3,2 | 59 | 12 | 131 | 89 | 0,3 |
| Th-230 | 1,5 | 14 | 17 | 15 | 332 | 68 | 733 | 495 | 1,5 |
| Th-231 | 8,1E+3 | 8,2E+3 | 1,1E+4 | 1,2E+4 | 8,8E+6 | 1,3E+6 | 4,9E+5 | 9,2E+5 | 8,1E+3 |
| Th-232 | 0,9 | 9 | 11 | 10 | 186 | 38 | 410 | 277 | 0,9 |
| Th-nat | 0,3 | 2,6 | 3,3 | 2,9 | 66 | 13 | 139 | 97 | 0,3 |
| Th-234 | 1,0E+3 | 1,6E+3 | 1,7E+3 | 1,6E+3 | 5,9E+5 | 1,2E+5 | 3,1E+5 | 4,2E+5 | 1,0E+3 |
| Ti-44 | 111 | 111 | 244 | 305 | 6,5E+4 | 9,6E+3 | 3,4E+3 | 6,3E+3 | 111 |
| Tl-200 | 211 | 209 | 497 | 680 | 2,7E+5 | 2,9E+4 | 6,0E+3 | 1,2E+4 | 209 |
| Tl-201 | 2,9E+3 | 2,9E+3 | 6,4E+3 | 8,5E+3 | 3,9E+6 | 4,2E+5 | 8,8E+4 | 1,7E+5 | 2,9E+3 |
| Tl-202 | 559 | 554 | 1,3E+3 | 1,7E+3 | 7,3E+5 | 7,8E+4 | 1,6E+4 | 3,1E+4 | 554 |
| Tl-204 | 3,8E+3 | 4,1E+3 | 4,2E+3 | 4,2E+3 | 1,2E+7 | 2,3E+6 | 5,7E+6 | 7,9E+6 | 3,8E+3 |
| Tm-167 | 1,3E+3 | 1,3E+3 | 2,3E+3 | 2,6E+3 | 1,5E+6 | 2,0E+5 | 5,2E+4 | 9,8E+4 | 1,3E+3 |
| Tm-170 | 1,7E+3 | 3,5E+3 | 3,6E+3 | 3,6E+3 | 6,6E+5 | 1,3E+5 | 7,3E+5 | 7,3E+5 | 1,7E+3 |
| Tm-171 | 1,2E+4 | 4,8E+4 | 5,2E+4 | 5,0E+4 | 3,3E+6 | 6,7E+5 | 4,4E+6 | 4,0E+6 | 1,2E+4 |
| U-230 | 1,3 | 15 | 21 | 17 | 290 | 59 | 638 | 432 | 1,3 |
| U-232 | 0,3 | 3,0 | 3,9 | 3,3 | 61 | 12 | 132 | 91 | 0,3 |
| U-233 | 2,3 | 25 | 33 | 28 | 485 | 99 | 1,1E+3 | 723 | 2,3 |
| U-234 | 2,3 | 26 | 34 | 28 | 495 | 101 | 1,1E+3 | 738 | 2,3 |
| U-235 | 2,6 | 28 | 37 | 31 | 547 | 111 | 1,2E+3 | 809 | 2,6 |
| U-236 | 2,5 | 28 | 37 | 31 | 535 | 109 | 1,2E+3 | 797 | 2,5 |
| U-238 | 2,7 | 29 | 39 | 32 | 581 | 118 | 1,3E+3 | 865 | 2,7 |
| U-nat | 0,5 | 3,4 | 3,9 | 3,6 | 119 | 24 | 248 | 173 | 0,5 |
| V-48 | 93 | 92 | 209 | 278 | 1,2E+5 | 1,3E+4 | 2,7E+3 | 5,2E+3 | 92 |
| V-49 | 2,6E+5 | 4,1E+5 | 4,1E+5 | 4,1E+5 | 1,4E+8 | 2,8E+7 | 3,0E+8 | 2,0E+8 | 2,6E+5 |
| W-178 | 2,2E+3 | 2,1E+3 | 4,8E+3 | 6,4E+3 | 2,8E+6 | 3,1E+5 | 6,4E+4 | 1,2E+5 | 2,1E+3 |
| W-181 | 6,2E+3 | 6,2E+3 | 1,4E+4 | 1,8E+4 | 8,2E+6 | 8,9E+5 | 1,9E+5 | 3,6E+5 | 6,2E+3 |
| W-185 | 9,8E+3 | 1,0E+4 | 1,1E+4 | 1,0E+4 | 3,9E+7 | 7,8E+6 | 6,7E+7 | 5,3E+7 | 9,8E+3 |
| W-187 | 512 | 508 | 1,1E+3 | 1,4E+3 | 7,1E+5 | 7,7E+4 | 1,6E+4 | 3,1E+4 | 508 |
| W-188 | 1,4E+3 | 1,4E+3 | 1,7E+3 | 1,7E+3 | 2,5E+6 | 3,7E+5 | 1,3E+5 | 2,5E+5 | 1,4E+3 |
| Y-87 | 336 | 333 | 778 | 1,1E+3 | 4,3E+5 | 4,6E+4 | 9,7E+3 | 1,9E+4 | 333 |
| Y-88 | 110 | 109 | 256 | 347 | 1,3E+5 | 1,4E+4 | 3,1E+3 | 6,0E+3 | 109 |
| Y-90 | 1,8E+3 | 2,0E+3 | 2,0E+3 | 2,0E+3 | 3,3E+6 | 6,8E+5 | 7,3E+6 | 5,0E+6 | 1,8E+3 |
| Y-91 | 1,3E+3 | 2,2E+3 | 2,2E+3 | 2,2E+3 | 6,5E+5 | 1,3E+5 | 8,8E+5 | 8,0E+5 | 1,3E+3 |
| Y-91m | 500 | 495 | 1,2E+3 | 1,6E+3 | 6,5E+5 | 6,9E+4 | 1,4E+4 | 2,7E+4 | 495 |
| Y-92 | 637 | 634 | 952 | 1,1E+3 | 1,4E+6 | 1,5E+5 | 3,2E+4 | 6,0E+4 | 634 |
| Y-93 | 1,3E+3 | 1,3E+3 | 1,7E+3 | 1,8E+3 | 3,1E+6 | 3,8E+5 | 9,3E+4 | 1,8E+5 | 1,3E+3 |
| Yb-169 | 746 | 743 | 1,5E+3 | 1,9E+3 | 7,0E+5 | 9,1E+4 | 2,4E+4 | 4,6E+4 | 743 |
| Yb-175 | 4,1E+3 | 4,1E+3 | 6,4E+3 | 7,0E+3 | 4,0E+6 | 5,7E+5 | 1,9E+5 | 3,6E+5 | 4,1E+3 |
| Zn-65 | 406 | 404 | 792 | 967 | 5,0E+5 | 5,9E+4 | 1,4E+4 | 2,6E+4 | 404 |
| Zn-69 | 8,3E+3 | 8,4E+3 | 8,4E+3 | 8,4E+3 | 1,7E+8 | 3,4E+7 | 2,8E+8 | 2,2E+8 | 8,3E+3 |
| Zn-69m | 622 | 617 | 1,4E+3 | 1,9E+3 | 7,9E+5 | 8,6E+4 | 1,8E+4 | 3,5E+4 | 617 |
| Zr-88 | 676 | 672 | 1,6E+3 | 2,1E+3 | 6,0E+5 | 7,6E+4 | 2,0E+4 | 3,7E+4 | 672 |
| Zr-93 | 2,0E+3 | 1,4E+4 | 1,6E+4 | 1,5E+4 | 4,7E+5 | 9,5E+4 | 1,0E+6 | 6,9E+5 | 2,0E+3 |
| Zr-95 | 348 | 346 | 783 | 1,0E+3 | 3,2E+5 | 4,0E+4 | 1,0E+4 | 2,0E+4 | 346 |
| Zr-97 | 164 | 163 | 356 | 464 | 2,2E+5 | 2,4E+4 | 5,0E+3 | 9,6E+3 | 163 |

# RADIOLOGICAL WORK MANAGEMENT ASPECTS INFLUENCING DOSE REDUCTION AT THE IGNALINA NPP DURING OUTAGES AND COMING DECOMMISSIONING

**G. Klevinskas**
Radiation Protection Centre, Lithuania

## Introduction

Lithuania has few nuclear facilities. There are two non-upgradable nuclear reactors of RBMK-1 500 series [light water-cooled graphite-moderated channel type reactors, LWGR (RBMK)]. Also one on-site dry spent nuclear fuel interim storage facility is located at Ignalina NPP.[1] All these facilities are located in the north-eastern part of Lithuania, near the borders with Belarus and Latvia, 160 km distance from the capital city Vilnius, on the banks of the lake Druksiai. The first Unit of Ignalina NPP went into operation at the end of 1983, the second Unit in August 1987.

Lithuania signed a Nuclear Safety Grant Agreement in 1994 stating, inter alia, the commitment of the Republic of Lithuania to close both units of the Ignalina NPP at the time of so called "gap closure" and therefore not to use the designed technical lifetime of the reactor units by re-channeling the fuel channels.

The decision to shut down the Unit 1 is already taken by law, and it is planned to stop the Unit 1 before the 1 January 2005. In November 2002 the Government of Lithuania adopted the decision which states that preferred option for the decommissioning of Unit 1 is the immediate dismantling. It means that the process of dismantling of the Unit 1 will start right after shutdown of the reactor and the process will continue at least for 30 years. From the radiological point of view, this process can be treated as big extended outage of the unit with constantly changing working environment that might cause high individual and collective doses to the workers.

1. The heat cycle of the Ignalina NPP reactors is identical to the Boiling Water Reactor (BWR) cycle extensively used throughout the world, and is analogous to the cycle of thermal generating stations. However, compared to BWRs used in Western power plants, the Ignalina NPP and other plants with the RBMK-type reactors have a number of unique features. The Ignalina NPP uses an RBMK-type channelised reactor. This means that each nuclear fuel assembly bank is located in a separately cooled fuel channel (pressure tube). There are a total of 1 661 of such channels and the cooling water flow rate must be equally divided among associated feeder pipes. After crossing the core, these pipes are brought together to feed the steam-water mixture to separator drums. The nuclear fuel assemblies of the Ignalina NPP are changed without shutting down the reactor. This is possible only for channel type reactors. It is possible to disconnect one of them at the time from the reactor cooling system, change the fuel assembly, and then reconnect the channel.

The draft of the Final Decommissioning Plan, as a part of the licensing requirements needed to start the decommissioning process, already exists. It has been preliminary estimated by the operator that during immediate dismantling of the Unit 1, the total exposure approximately will result in 35 manSv.

Due to unique construction, the occupational exposure results for the reactors of RBMK (LWGR) type are one of the highest if comparing with reactors of other types. The outages of both units each year usually last 2-4 months. The main “contributors” to the annual collective dose are the annual outages of the units. Keeping the occupational exposure in accordance with legal requirements, the operator needs to establish an effective radiological work management programme during the outage periods, which shall serve as a part of the radiation protection programme.

Both legal requirements and practical experiences applied at Ignalina NPP related to radiological work management aspects influencing dose reduction during outages and radiation protection measures planned to be applied during coming decommissioning of both units is discussed in the paper.

**Legislation**

The main laws and regulations of the Republic of Lithuania establishing radiation protection requirements are the Law on Radiation Protection (passed in 1999) [1], the Lithuanian Hygiene Standard HN 73:2001 “Basic Standards of Radiation Protection” [2] and HN 87:2002 “Radiation Protection in Nuclear Facilities” (hereinafter – HN 87:2002) [3] and other supplementary radiation protection legislation.

The HN 73:2001 establishes radiation protection requirements for practices, classifies the radiation workers, sets the dose limits that are in accordance with European Union requirements and international recommendations on radiation protection [4, 5]. The effective dose limit for the occupational exposure is 100 mSv in a consecutive 5 year period, subject to a maximum effective dose of 50 mSv in any single year [2]. Annual public dose constraint set for the operation and decommissioning of nuclear facilities, is 0.2 mSv [3].

The regulatory authority in the field of radiation protection is the Radiation Protection Centre, RPC. It co-ordinates the activities of executive and other bodies of public administration and local government in the field of radiation protection, exercises state supervision and control of radiation protection, including nuclear facilities, performs monitoring and expert examination of public exposure. The functions and responsibilities of the Radiation Protection Centre are described in Article 7 of the Law on Radiation Protection.

**Requirements for occupational radiation protection in nuclear facilities – radiation protection programme**

According to the requirements set out in [3], the radiation protection programme shall be established at the nuclear power plant. The radiation protection programme addresses following issues:

- classification of working areas and access control;

- local rules, measures of supervision of safety at work and order of organisation of work;
- procedures of monitoring of workplaces and individual monitoring of workers;
- individual protective equipment and procedures for their application;
- main premises, control systems for assurance of radiation protection;
- requirements for management of radioactive waste;
- radiation protection in the case of accidents;
- application of optimisation principle (ALARA) and measures on exposure reduction;
- programmes of health surveillance;
- compulsory training of workers and their instructions.

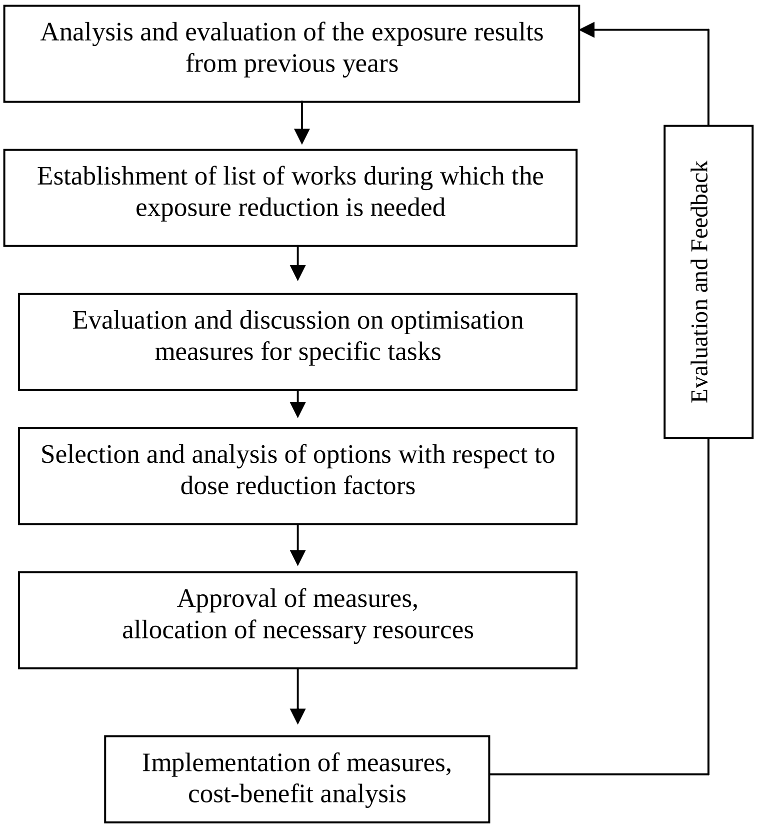
The radiation protection programme implemented at the Ignalina NPP basically conforms with the programme given in [6]. In accordance with the INPP management procedure "Radiological Protection", QA-2-005, in the frame of the INPP Radiation Protection Programme, a set of procedures was created.

**ALARA strategy at the Ignalina NPP. ALARA group and ALARA programme**

The process for implementing of the ALARA procedure at the Ignalina NPP is given in Figure 1.

The implementation of ALARA programme started at Ignalina NPP in 1996. The programme defines the procedure for exposure planning, definition and implementation of necessary measures and financial resources needed for exposure reduction. The establishment and implementation of the ALARA programme is required by [3]. Since 1998 the ALARA group is established at the plant. The main task of the group is to plan the occupational exposure doses and to establish necessary recommendations for the dose reduction. In order to effectively control the doses of personnel, the ALARA group plans the daily, monthly and annual doses. Planned annual doses with doses associated with specific tasks during the outages are established and agreed with the Radiation Protection Centre. The effectiveness of the ALARA programme is monitored by the ALARA group through the whole duration of the work year of the plant. The evolution of actual individual and collective doses is compared to the evolution of the predicted values. The results of the radiological protection measures implemented during the outages, results of occupational exposure are discussed, and possible ways and main areas for improvement of working conditions when performing specific jobs are evaluated and discussed in the regular meetings of all workshops, especially after outages. The results of the ALARA programme are addressed in reports generated after outages and sent to the Radiation Protection Centre and other regulatory authorities for evaluation.

**Figure. 1. The ALARA procedure at the Ignalina NPP**




Generally, the ALARA approach addresses the following basic parts described below:

a) Work organisation. Types of jobs, during their performance the staff receives high individual doses, are described in specific work description list. Mainly there are: works on metal control; repair works of the Main Forced Circulation Circuit; works on cutting of SNF rods, rods of additional absorbers; decontamination, treatment and removal of radioactive waste. At the end of each year, the INPP departments and outside organisations prepare the collective dose plan for each radiological work and present and discuss it with the representatives of ALARA group. The summarised dose plan for coming year is prepared by the RP Department and approved by the RPC. Specific jobs are performed by well-trained teams, whose members have enough experience in performing of complex tasks in the fields with high dose rates.

b) Training for personnel.

c) Improvement of working environment.

d) Perfection of technological processes.

e) Implementation of quality assurance programme.

f) Safety culture. Since 1997 the active development of safety culture among staff has begun. Safety culture indicators are used for timely prevention of hidden deficiencies and also in such cases when the positive trends towards increase of safety level occur. Six indicators of safety culture used at INPP are identified: Indicator 1 – Safety culture seminars; Indicator 2 – Recommendation of Safety Committee; Indicator 3 – Deficiencies identified during audits; Indicator 4 – Repeated events; Indicator 5 – Human error; Indicator 6 – Proposals on safety improvement. The evaluation of these indicators is carried out together with Safety culture assessment.

g) Evaluation and avoidance of influence of human factor. For the evaluation of influence of human factor as errors and incorrect operations of personnel, that can cause an unjustified exposure, necessary means such as barriers, locks, preventive signalling systems, posters and signs are used.

**Radiological work management approaches and means of reducing exposure during execution of specific task during outages**

***Work permits***

The planning of work to be undertaken in the controlled area where it is possible that levels of radiation or contamination may be significant is an important mean of keeping doses ALARA. This approach is taken into account if the tasks within the controlled area necessitate radiological precautions, and they are allowed only if work permits are issued. The work permits consist of specific measures to be implemented and applied by workers when they carry out the works with high dose rates and in complex working environment.

Following order for the filling up and approval of the work permit is applied:

- the health physicist investigates and estimates the radiation conditions within the premises (measures the dose rates, estimates the surface, air contamination levels, etc.);
- operational workers prepare the workplaces;
- if there is possibility for external, internal or air contamination, appropriate protective equipment to be used is described and appointed by the work permit;
- before the beginning of works, in order to perform them as short as possible, the team members are instructed how to perform particular jobs, how to optimise particular operations;
- work permits are issued and approved by the Shift Head of Work Safety Department.

Use of the work permits allows effectively manage radiological works and therefore serves a basis for exposure optimisation.

**Decontamination of systems**

Good practice used at Ignalina NPP during the outages giving positively results in reduction of the dose rates, is the decontamination of systems. An example of decontamination factors obtained after the flush out of the pipes of the main forced circulation circuit during the outage of Unit 2 in 2003 are presented in Table 1.

**Table 1**

| Room No. | Gamma dose rate, mSv/h | | Decontamination factor |
|---|---|---|---|
| | Before decontamination | After decontamination | |
| $117/_{1\text{-}7}$ | 1-400 | 0.6-1.2 | 1.7-330 |
| $409/_{1}$ | 50-400 | 2-2.3 | 25-170 |
| 213; $214/_{1,2}$; 215 | 1.5-100 | 0.2-1.1 | 7.5-90 |

**Installation of temporary radiation shielding**

All tasks subject to high dose levels are evaluated for the effectiveness of installing temporary radiation shielding. Lead blankets are mostly used. Temporary shielding is installed only in those cases, if the results of evaluation show that the averted dose will be higher than received during the installation of temporary shielding. Table 2 shows the examples of reduced dose rates after the installation of temporary radiation shielding during the repair of pipes Du-300 during the outage of Unit 2.

**Table 2**

| Room No. | Gamma dose rate, mSv/h | |
|---|---|---|
| | Without radiation shielding | With radiation shielding |
| $506/_{1,2}$, near the blind pipes | 5.0-2 000 | 5.0-45.0 |
| $208/_{1}$, near the joints 6,8 of headers of the emergency cooling system | 8.0-400 | 8.0-70 |

**Training of personnel**

A knowledge workforce is one of the fundamental elements in the radiation protection programme for the optimisation of protection and control of exposure. In addition to the general training, on all planned works the advanced training for personnel is organised. The procedure for self-checking and self-assessment has been improved owing to regular special training. Everyday, before the maintenance works, the conditions for carrying out of each work are discussed with workers during operational meetings. 240, 60 and 30 hours training in radiation protection is required before starting the work first time, for persons responsible for radiation protection and for workers accordingly. The frequency of training is 5 years. Specific ALARA questions are included in personnel training programme. In 2002, the educational book on radiation protection was introduced. The draft Lithuanian edition of the book is prepared. Training on mock-up of equipment is very intensively applied, which allows to obtain experience and to become personally "familiar" with the work before its execution in fields with high dose rates.

**Occupational exposure results during outages in 2003**

The outage of Unit 1 lasted 12/07/03-21/10/03, the outage of Unit 2 – 05/04/03-01/06/03. There was also one short unplanned outage of Unit 2 performed – from 1 to 6 of November 2003. As it has been planned by the operator at the end of 2002 and approved by the RPC, following activities have mostly contributed to the annual collective doses during the annual outages of Units 1 and 2 in 2003:

- replacement of reactor fuel (technological) channels;
- maintenance of primary system pipes Du-300 (d=300 mm);
- maintenance of primary system pipes Du-800 (d=800 mm);
- control of metal;
- replacement of spherical valves used for measuring of outlet water flow (SHADR-32M) and replacement of removable parts of regulation valves;
- repair of pipes of the main circulation circuit, reactor emergency cooling system, blow-off and cooling system of the reactor;
- insulation works;
- installation of temporary shielding.

The radiation works during the normal operation of both units (between the outages) usually contributed not more than 20 percent to the annual collective dose. This tendency is observed for a several years.

The planned collective dose for plant personnel was 7.59 manSv, for contractors – 2.57 manSv, actual collective dose was 6.66 manSv and 1.88 manSv respectively (less than 12% and 27% accordingly than planned). In 2003 the total number of 4 458 employees was individually monitored for radiation, including contractors and visitors. The average individual dose was 2.25 mSv for plant personnel and 1.25 mSv for contractors.

There were following achievements gained in radiation protection in 2002-2003 at Ignalina NPP:

- New whole body counter "ACCUSCAN 2260-G2KG" was putted into operation. In 2003 the monitoring of internal exposure was carried out for 2 659 workers. There was no over-exposure, exceeding limits, detected.
- New system for training of personnel in the field of radiation protection has been introduced. With support from Sweden the educational book "Radiation Protection" in Russian and Lithuanian is issued. Amended programmes for training and checking of knowledge of personnel have been applied.
- Replacement of equipment for measurement of radioactive contamination at the exits from controlled area (RZB-04 have been replaced to new RTM-860, PPM-1, PMW-3e).
- Set of dosimetric equipment is renewed (50 items of RAD-62-, 4 items of TELETECTOR, 2 items of FH 40 G with detector FHT 752).

Some of the main measures for optimisation of exposure implemented during the outages in 2002-2003 were as following:

| | |
|---|---|
| *Replacement of fuel (technological) channels* | Installation and training on mock-up.<br>Purchase of equipment and putting into operation of non cable connection<br>Installation of equipment for removal of filing.<br>Preparation and implementation of additional equipment for replacement of fuel channel. |
| *Maintenance of pipes Du-300, Du-800* | Training of personnel from Centralised Maintenance Workshop on replacement of welded seams on the mock-up.<br>Purchase of equipment and training of welders on welding of pipes with equipment of type VAS 120.<br>Purchase of equipment for cutting and replacement of parts of pipes Du-800.<br><br>***Training of personnel on repair technology of pipes Du-300 on the mock-up***<br><br>Application of automatic welding for pipes Du-300. |
| *Replacement of spherical valves used for measuring of outlet water flow (SHADR-32M) and replacement of removable parts of regulation valves* | Training of personnel on repair technology on the mock-up. |

## Planned decommissioning activities and ALARA approach

With reference to in the middle of 2002 amended National Energy Strategy and other related legislation, in November 2002 the Government of Lithuania adopted the decision stating that preferred option for the decommissioning of Unit 1 is the immediate dismantling. The process of dismantling of the Unit 1 will continue at least 30 years. From the radiological point of view this process can be treated as big extended outage with constantly changing working environment that might cause high individual and collective doses to the workers.

The draft of the Final Decommissioning Plan, as a part of the licensing requirements needed to start the decommissioning process, already exist. It was preliminary shown by the operator that during immediate dismantling of the Unit 1 the total exposure will result in 35 manSv [7].

Therefore, for the operator (Ignalina NPP) it will be important to plan the radiological works well in advance before the dismantling of the equipment will start, predict occupational exposure results during particular dismantling, decontamination and other operations, as well as to show for the regulatory authorities that doses to the workers are kept ALARA. The implementation of the radiation protection and ALARA programme during the decommissioning will be a main issue of concern in order to keep the occupational doses within the established limits. In case of immediate dismantling, dose commitment to staff dedicated to core and primary circuit equipment dismantling is of concern.

**Preparatory activities and interfaces with ALARA commitment during the decommissioning**

The compliance with the ALARA policy objectives for the plant personnel basically relies on 2 key issues that, actually, are already implemented during the routine operation of INPP:

- the preparation of the tasks to be carried out in the controlled area with a high level of details, and carrying out of mitigation measures;
- the monitoring of the individual and collective doses during the execution of the tasks and the implementation of corrective actions in case of violation of the pre-established ALARA objectives.

These key issues will still remain applicable during dismantling works.

Keeping the personnel exposure in accordance with ALARA principle during decommissioning will require:

a) careful engineering, technical, administrative preparations of the activities to be carried out;

b) the monitoring of their implementation in the controlled area and implementation of corrective actions in case of individual and collective dose exceed the ALARA objectives.

The preparatory activities involve all the engineering, technical and administrative preparatory tasks to be conducted prior to carrying out activities in the controlled area such as:

a) authorisation of works in the controlled area by work permits;

b) establishing the dose maps in the to be accessed areas;

c) carrying out computer simulations of the activities to be performed in the controlled area. These simulations enable, for example, to evidence specific tasks or sub-tasks that leads to high individual and collective exposures;

d) assessing, accordingly, the need to implement countermeasures to reduce the background: decontamination, installation of temporary shielding, use of remotely operated equipment, additional ventilation capacities, additional confinement zones, etc.;

e) careful review of the lessons to be learned from previous similar activities already carried out either at INPP or at other NPPs and, when relevant, adaptation or optimisation of the operation procedures;

f) definition of dosimetric objectives;

g) instructions and training of the operators;

h) checklists of the specific tools and equipment (including additional radiation monitoring equipment, needed in the controlled area);

i) checklists of the equipment to be consigned (isolated).

In order to comply with the strategic ALARA objectives, Ignalina NPP is going to implement complementary tools in the frame of the decommissioning project, such as e.g. computer software (LLWAA DECOM software, VISIPLAN 3D ALARA software, etc.). Experience gained in radiological works management during the outages in normal operation of the units will be also considered, as it was discussed above.

**Conclusions**

A set of procedures, forming of the radiation protection programme, was created and is effectively implemented at INPP. Good radiological work management procedures applied during the outages give successful results in reduction of occupational exposure.

Keeping the personnel exposure in accordance with ALARA principle during the decommissioning of Ignalina NPP will require careful engineering, technical, administrative preparations of the activities to be carried out. Positive experience gained in optimisation of protection during the normal operation of the Ignalina NPP will be taken into account.

**References**


1. Law on Radiation Protection of the Republic of Lithuania, No. VIII-1019, (1999).
2. Lithuanian Hygiene Standard HN 73: *Basic Standards of Radiation Protection*, (2001).
3. Lithuanian Hygiene Standard HN 87: *Radiation Protection in Nuclear Facilities*, (2002).
4. IAEA (1996), *International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources*, Safety Series No.115, Vienna.
5. EURATOM (1996), Council Directive 96/29/EURATOM of 13 May 1996 laying down Basic Safety Standards for the protection of the health of workers and the general public against the dangers arising from ionizing radiation, No. L 159, Vol. 39.
6. IAEA (2002), *Radiation Protection and Radioactive Waste Management in the Operation of Nuclear Power Plants, Safety Standards Series,* Safety Guide No. NS-G-2.7, Vienna.
7. Final Decommissioning Plan. Draft 4 A1.4/FDP/0004.

## ALARA IMPLEMENTATION AT UKRAINIAN NPPS

**T. Lisova**
Nuclear Energy Department, Ministry of Fuel and Energy of Ukraine

**Y. Roshchyn**
National Nuclear Energy Generation Company "Energoatom, Ukraine

### Introduction

Work Management Principles utilisation in the nuclear industry leads to harmful influence decreasing that means decreasing occupational dose exposure. The decrease of occupational dose exposure is often reached by means of decreasing the number of workers in the control zone, reducing the working time, reducing the amount of error correction work during different stages of operation. At that the objective of costs reduction as well as outage time minimisation can be reached that leads to increasing electricity production (generation).

ALARA activity is based on following principles:

- Any exposure will be justified if a prospective benefit is higher than a potential risk of exposure;
- exposure must be held at a minimal possible level, in view of all social and economic conditions;
- exposure must be restricted by limits established by rules and instructions in order to minimise the exposure risk.

### ALARA implementation in the National Nuclear Energy Generation Company "EnergoAtom"

Following the above principles the utility organisation NNEGC "Energoatom" for 5 years have been carrying out systemic work in the field of radiation protection and radiation safety. In 2000-2001 Radiation Protection groups/ALARA groups were created at all Ukrainian NPPs. Regulations (sets of rules, instructions) were issued at each NPP in order to manage radiation-dangerous jobs and manage radiation protection in general.

**ALARA groups functions**

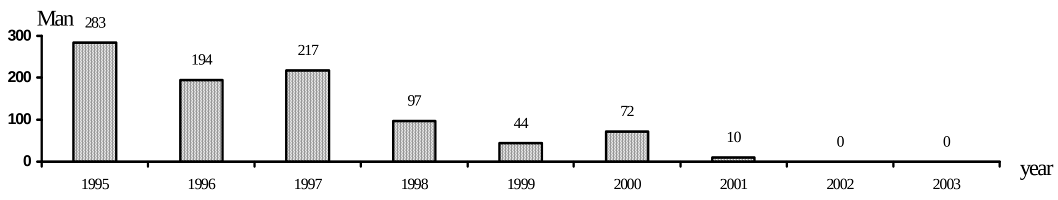
- Carrying out analysis and work planning with the purpose of achieving the highest possible personnel dose reduction and not exceeding the individual effective dose exposure of more than 20mSv/year.
- Putting into practice such organisation of labour and method of work performance in “a stringent operation condition zone” when exceeding dose limits is not practically possible as prescribed by a job instructions for these procedures; and when all personnel in a work management link (head of division – head of section – foreman – superintendent of work – member of crew) are understanding and realising their personal responsibilities and duties while performing such particular jobs.
- ALARA programme acceptance and review.
- Establishing and approval of annual exposure indicators.
- Preparation, consideration and approval of annual and prospective measures to decrease exposure and increase radiation protection level.
- Consideration during their meetings of ALARA programme performance, collective dose level and decision making to improve the programme’s efficiency.
- Preparing information (data) in order to approve the doses planned for NPPs as a whole for a year, for a planned unit outage, for separate divisions, and if necessary – for the most dangerous jobs.
- Analysis of repair documents, job programmes, safety aids, maintenance regulation with regards to adequacy of radiation protection measures, measures performance control.
- Analysis of prospective works during unit outage, radioactive-dangerous jobs specifying, outage documentation checking for the purpose of organisational and technical evolutions to ensure not exceeding of the planned dose exposure for these jobs and development of measures for decreasing dose exposure.
- Analysis of dose exposure for the accounting period ( quarterly, if necessary – monthly) for units, divisions, for separate dose-value operations; after that – development of recommendations for decreasing the doses on the basis of the analysis.
- Work planning control performance for a unit outage, integrated operational schedule for one, for daily and weekly tasks. At that all outage’ papers must be considered (agreed) with the expert (head) of health physics division.
- Participation in the newly performed radiation dangerous jobs.

As a result of the activity carried out and comprehensive approach during the outage’s work planning the forecasting based on the previous works analysis of division’s collective dose exposure has been put into practice. Division heads were made responsible for workers’ individual doses; the list of the organisation measures to decrease the dose input is being made for each planned outage.

**Individual and collective dose analysis in the NNEGC "EnergoAtom"**

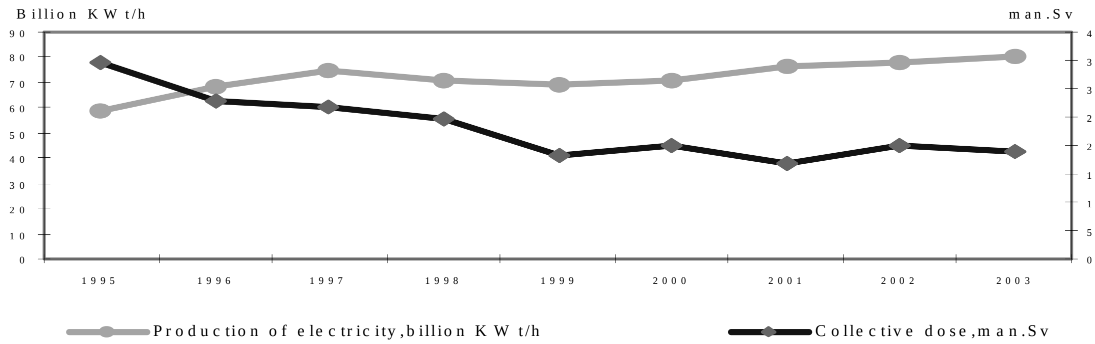
During the recent 9 years, the annual collective dose at the Company's NPPs has had the tendency to decrease.

**Graph 1. The number of WWER personnel having taken the external dose more than 20 mSv per year during 1995-2003**




As we can see from the Graph 1 during 2002 and 2003 in the Company no event of exceeding the main limit of individual dose of exposure – 20 mSv per year was recorded.

**Graph 2. The trends of collective dose change and amount of electricity production in NNEGC "EnergoAtom" during 1995-2003**

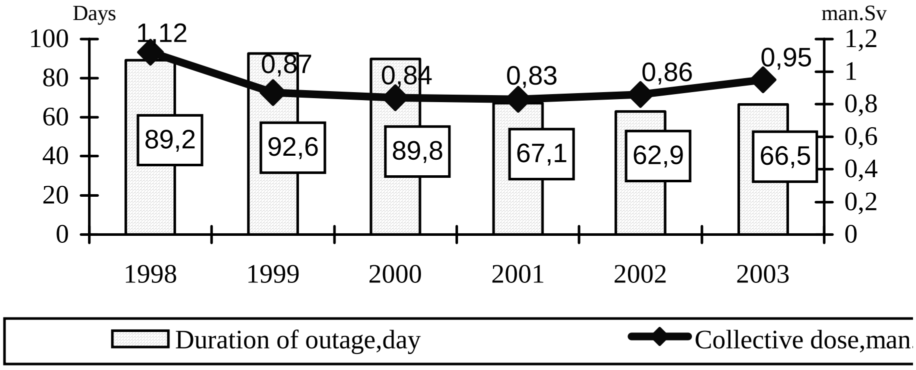



As we can see from Graph 2 that under a stable tendency of electricity production increase (due to a load capacity factor) by the Company during recent nine years the total value of collective dose of NPP's personnel was steadily decreasing up to 1999 and from this period has remained at the same level. So, the total collective dose exposure of NPP's personnel in 2003 was 18,8 man.Sv, that less by 1,2 man.Sv in comparison with 2002.

In Graph 3 outage personnel collective exposure doses are presented and average outage time for one unit.

The data analysis shown in the Graph 3, indicates that the curve reflecting the level of outage personnel collective dose is identical to the curve of the total personnel collective dose of the Company for the recent years (see Graph 2). Beginning with 1999 the level of outage dose was decreasing and for recent three years has not essentially changed. In 2003 due to the outage time increase per unit the collective dose per unit increase accordingly. Taking into account substantial input of outage doses (70-80%) into the total collective dose beginning with 2002, in NNEGC “Energoatom” outage doses have been planned for each unit outage.

**Graph 3. Collective exposure dose and duration of outage per unit in NNEGC “EnergoAtom” for a period from 1998 to 2003**




In 2002 into NNEGC “Energoatom” completed the development of the normative document (guide) “Methodological Guidelines for the Collective Personnel Exposure Dose Analysis during Planned Outage and Equipment Maintenance Activities at NPPs” with the purpose of calculation unification of work-consuming and dose-consuming tasks for the repair and maintenance works for separate systems and separate jobs. Development of this document was based on current experience at NPPs regarding such accounting in compliance with international recommendations (adapted with ISOE).

This document entered into force by orders of both the Ministry of Fuel and Energy of Ukraine and the Company. Implementation of the “Methodological Guidelines…” must establish and promote high quality comparative analysis on the basis of experience exchange and liven up the work decreasing occupational exposure that will finally lead to improving radiation safety at NPPs.

NNEGC Board have created a section as a part of Scientific-Technical Council called “Radiation Protection and Radioactive Waste Management” where the most important issues for the different ways of activity are considered. By the end of 2003 “Programme on Decreasing of NPP Staff Exposure from 2004 to 2008” was developed end entered into force in which organisation and technical evolutions at each NPP and needed funds for its execution are specified.

**Currently at each NPP**

- The “Lists of radiation-dangerous jobs, operational and Repair Procedures” have been developed.
- Forecasting of collective dose exposure for a unit outage has been implemented.

- Reviewing of on-going programmes of radiation – dangerous works has started in concordance with “Lists of radiation-dangerous jobs, operational and repair procedures” with purpose to ensure radiation protection.
- Engineering groups (or an appointed worker) for performing the analysis and processing the information concerning division collective dose changes and major repair activities have been organised.

In that way by NNEGC “Energoatom” and its NPPs (on-site) a sufficient range of activities in the field of ensuring radiation protection and radiation safety is performed.

**Problems**

However at present several problems which prevent correct practical and organisational Radiation Protection aspects implementation are not resolved. The major problems involve creation/ review of normative – methodological documents of the highest level and bringing the existing documents to conformity.

Currently, the same common problems connected with personnel exposure exist for all Ukraine’s NPPs. Thus, the lack of modern electronic dosimeters at NPPs leads to additional mistakes during exposure dose determination and man-hour determination for separate radiation-dangerous activities.

At present a limited quantity of electronic dosimeters is available at Rivne NPP, not so long ago 300 MGP dosimeters were received at South-Ukraine NPP, the rest of NAEK “EnergoAtom” NPPs have not got them in spite the decision taken about their purchase. In general, this is connected with their high cost and other financial problems of the Company.

**Conclusions**

Introduction of automatic systems of metal quality examination and control, efficient remote methods of decontamination, remote visual control means (television systems) utilisation, steam generators and high-level equipment tightness control systems – these are the main means to reduce the quantity of the personnel having doses approaching the permissible and collective doses reduction.

Resolution of these problems will allow NPP radiation protection services respond adequately and in proper time to processes of collective occupational dose formation during radiation dangerous activities in time of corrective maintenance, outage. Organisational measures directed at exposure reduction is not enough. That is why at this stage of ALARA principle introduction the attention should be focused on technical aspects of the problem solution.

**ALARA principle implementation during activities at the Chernobyl state specialised enterprise and at the “Shelter”**

As you know, Chernobyl NPP was shut down on 15 December 2000. The Chernobyl state specialised enterprise was created on its basis. The activity of this enterprise is directed at Chernobyl

NPP decommissioning and Shelter Implementation Plan (transformation of the "Shelter" facility into ecologically safe system). In order to solve these problems "The Integrated Programme of Radioactive Waste Management during Chernobyl NPP Decommissioning and Transformation of the "Shelter" Facility into Ecologically Safe System" was developed.

The main objectives of this programme are as follows:

- Preventing radioactive waste release into environment at different phases of radioactive waste management.
- Minimisation of the amount of a radioactive waste arising at Chernobyl NPP and the "Shelter".
- Arisen radioactive waste reprocessing.
- Personnel exposure dose reduction during radioactive waste management.

The programme objectives have to be accomplished by means of:

- ALARA principle implementation at all stages of radioactive waste management.
- All reconstruction and modernisation activities analysis in order to minimise radioactive waste arising.
- Development and implementation of new radioactive waste reprocessing technological processes.
- Personnel training in technologies of modern radioactive waste management.
- Uniform radioactive waste accounting system development.

**Optimisation principle utilisation during the "Shelter Implementation Plan" projects development**

Currently the "Shelter" Radiation Protection Programme has been developed and is being agreed with the regulatory bodies. In the above programme the ALARA principle implementation for ensuring the radiation protection level during "Shelter" activities programmes development could be shown.

It is necessary to systematise the basic information during "Shelter" activities programmes development and if needed to perform additional research. As a result the information has to be obtained concerning:

- *Radiation sources identification and location*. To assess the level of potential exposure it is necessary to identify these radiation sources to estimate probable accidents and connected probable potential exposure during the pre-design research stage.
- *Personnel movement hindrances*. The hindrances have to be identified, defined or eliminated. These provisions have to be made during the working schedule and technological maps development. This approach envisages not only identification movement routes but measures of their development before the beginning of activities. Such activities have to be

defined as preparation activities and should allow to minimise the doses absorbed by the personnel having access to places where works are performed. Besides, places which need shielding have to be defined on the maps. The shielding has to be foreseen according to the number of passages, the number of workers passing and prevented exposure dose.

- *Personnel movement routes*. The routes are to be properly defined and illuminated. During instruction activities the personnel has to learn properly not only the main routes but accident evacuation routes as well. Thus, different movement routes have to be foreseen in the design documentation; the main route as well as the additional one.
- *Radiation situation in the places where works are being performed*. Maps of $\gamma$ – radiation exposure dose capacity distribution, $\alpha$ и $\beta$ – particles surface contamination density and bulk concentration of aerosols in the air have to be obtained.
- *Work performance zones conditions concerning their illumination, space closure, etc*. Information connected with meeting the industry safety requirements influences greatly the optimisation of the work performance process. Taking into account the necessity of preparation activities connected with illumination system creation also influences the collective exposure dose during work performance. Under normal illumination conditions the personnel will perform the work faster and the exposure dose will be smaller. In the work performance plan it is necessary to take such technical decisions concerning power supply and illumination system creation so that preparation of their installation in radiation dangerous places could take minimum time, and the major works could be performed in the clean zone. Space closure also plays an important role in exposure dose optimisation. In such areas it is necessary to use such technologies, to use such number of people so that rigging and equipment could not hinder work performance; the number of people has to be minimised.

According to the information obtained it is necessary to perform the analysis of suggested technical decisions. The decisions themselves have to foresee several implementation options. Among the suggested options, the one which could allow personnel collective dose reduction ought to be chosen.

**References**


1. Law of Ukraine “*Nuclear Energy Utilization and Radiation Safety About*”, (1995).
2. Law of Ukraine “*Radioactive Waste Management About*”, (1995).
3. Law of Ukraine “*Protection of Man from Acting Ionizing Radiation About*”, (1998).
4. Decree by Board of Governors of Ukraine “*Measures for Transformation of* “*Shelter*” to the ecological safety system”, (N1561, 1996).
5. Ukrainian Hygiene Standard “*Radiation Safety Standard of Ukraine*”, (1997).
6. Ukrainian Hygiene Standard “*Radiation Safety Standard of Ukraine/Supplement: Protection against Potential Exposure*”, (2000).

# DEVELOPMENT OF WORKING METHODS USED INSIDE REACTOR PRESSURE VESSEL AT OSKARSHAMN FROM THE RADIATION PROTECTION POINT OF VIEW

**C. Solstrand**
OKG Aktiebolag, Sweden

## Introduction

When performing maintenance and repair work in the beginning of 1970's conventional work tools and working methods mostly were used. The focus was on how to protect workers sufficiently in a proper way to keep the doses ALARA.

During the last years the focus has turned more towards construction of special work tools in order to minimise personnel doses without any special arrangements for radiation shielding.

Three examples will be presented to show that the optimisation of radiation protection can lead to the development of work tools and working methods, reducing the doses, time and money.

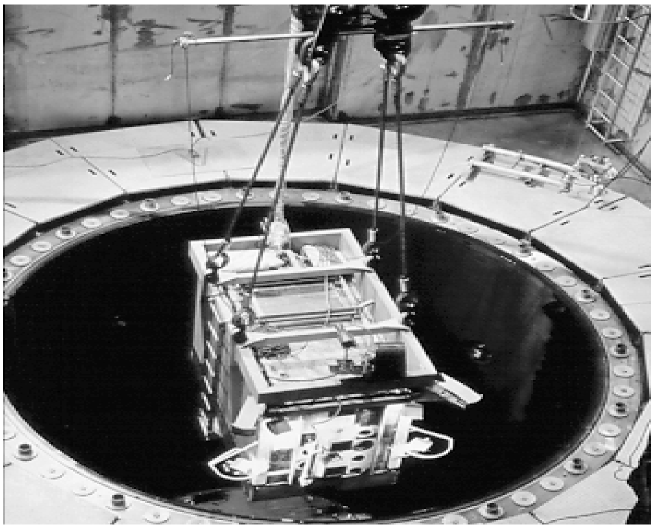
## Replacement of the feed water ring at Oskarshamn 1

In 1974 the feed water ring at Oskarshamn 1 was replaced because of cracks found in the material. Initially all fuel was removed from the rector pressure vessel to the fuel pond. The water level in the RPV was kept 50 cm above the core grid as a radiation shielding. Though the dose rate was too high to perform the planned job. Complementary radiation shielding had to be provided in order to be able to perform the repair work in an acceptable way from the radiation protection point of view.

Therefore a lead box was constructed and manufactured of lead bricks with a thickness of 10 cm. In the lead box there was space for two workers. It was equipped with a window made of lead glass, holes for the arms and over pressure ventilation. The lead box was hanging in the reactor hall crane and moved up and down in the RPV. Using conventional work tools the feed water ring was cut away and four new feed water spargers were welded in place.

The work was performed on three shifts and a total of 107 workers were during shorter or longer times engaged in the work. Totally 55 workers took part in the work from the lead box. The total collective dose for the job was 440 man.mSv of which only 50 man.mSv received in the lead box. The rest of the dose was received in the reactor hall. The highest individual dose was 20 mSv and the highest hand dose was 34 mSv.

**Picture 1. The lead box hanging in the reactor hall crane**



## In-service inspections of the bottom nozzles of the RPV at Oskarshamn 1

In 1994 in-service inspections of the bottom nozzles of the RPV at Oskarshamn 1 was performed. In order to be able to perform this job using conventional equipment a system decontamination of the RPV was performed. The system decontamination was very successful and the contamination levels and the dose rates at the bottom of the RPV were fairly low. The contamination levels varied between 500 and 10 000 kBq/m$^2$ and the dose rate level was about 0,04 mSv/h.

The workers were wearing only airflow protective suits in order to prevent inhalation of any particles and to minimise the spread of contamination. Anyhow the system decontamination cost a lot of money and time but the inspections could be performed by conventional equipment. The collective dose for the in-service inspection of the bottom nozzles of the RPV was 115 man.mSv.

**Picture 2. Picture of the bottom nozzles of the reactor pressure vessel**

**Repair of water level measuring nozzles of the RPV at Oskarshamn 2**

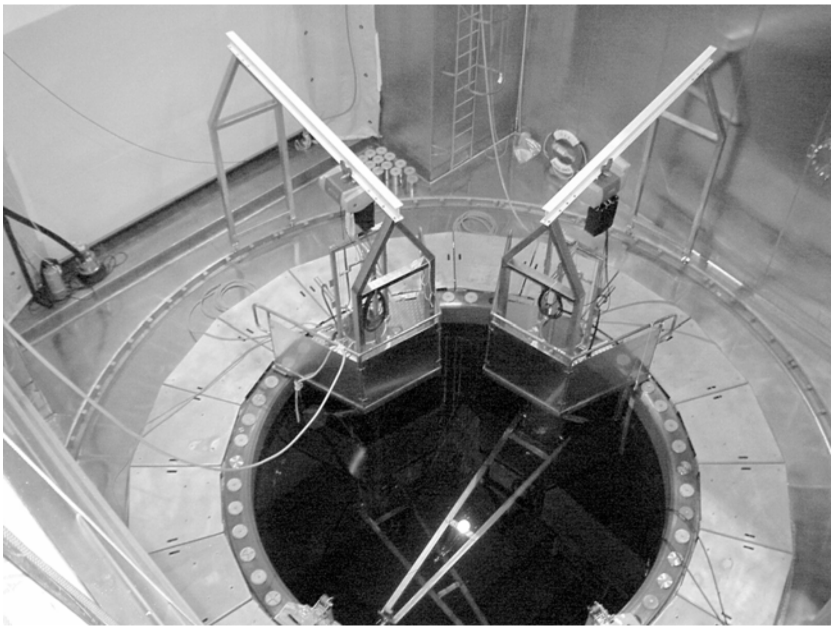
During the outage period of Oskarshamn 2 in 2003 in-service inspection of the water level measuring nozzles (LMN) of the RPV was performed. Several indications were detected. In 8 indications of totally 15 LMN:s cracks were found and measures had to be taken. Because the results were expected a repair method had been carried out during the two months before the outage.

The LMN:s are situated on three different levels, 4, 6 and 14 m down from the RPV flange. Repair work was performed trough special designed shafts. These were attached to the flange of the RPV and went down to the LMN:s and attached to the wall of the RPV by a gasket covering the LMN:s. All water was evacuated from the shafts.

The diameter of the shafts was about 50 cm. Special equipments for milling, welding and testing were constructed to fit into these shafts. The function was tested on special designed mock-ups. Cameras were mounted on the equipment to make it possible to locate, observe and operate the equipment from TV-monitors. The equipments were hanging in wires in the shafts and operated electrically and pneumatically from the floor of the reactor hall.

The repair work started with milling of the flange of the LMN, after that welding and finally testing. It was possible to perform repair work parallel in two separate shafts.

**Picture 3. Picture of the two work shafts attached to the flange of the reactor pressure vessel**



The repair work lasted for 30 days. The dose rates on the working area at the RPV flange varied between 0,07 and 0,2 mSv/h. The total collective dose of this job was only 47 man.mSv.

## Discussion and conclusions

The two first presented jobs are examples on works that have been performed using conventional work tools and trying to find radiation protection measures to keep the personnel doses ALARA. Usually this means that some kind of radiation protection actions have to be adjusted to the actual situation.

In the third example the focus was on the work tools as well as on the dose reduction. Special equipment was constructed to solve the upcoming problem and to reduce the need for personnel protection and radiation shielding. The job was performed in a relatively short time and the resulting collective dose was low. If conventional work tools had been used, together with radiation shielding or decontamination, the collective dose would certainly have been higher. Surely it would also have taken longer time and cost more money.

Thanks to the aspects of radiation protection the work tools, for this type of work in the RPV, have gone through a development resulting in both lower doses and in saved time and money.

## SESSION V

# CONTROL OF OCCUPATIONAL EXPOSURE WHEN WORKING WITHIN A REACTOR CONTAINMENT BUILDING AT POWER

**M.P. Lunn**
British Energy Generation Ltd., Sizewell B Power Station, United Kingdom

## Introduction

Sizewell B is a 1200 MW, 4 Loop Westinghouse-designed Pressurised Water Reactor, owned and operated by the private utility, British Energy. In the extremely competitive United Kingdom electricity market, where wholesale electricity prices have fallen as low as €11 per MWh, generators are under intense pressure to reduce their costs. Sizewell B has attempted to reduce costs by achieving shorter refuelling outage durations. One technique has been to maximise the scope of work performed whilst at power, including work inside the reactor containment building. This paper describes the radiological challenges presented by a routine containment entry programme and the techniques used to manage doses.

### *Radiological hazards at power*

The information on the radiological conditions come primarily from surveys conducted during station commissioning and on subsequent containment entries, and also from Monte Carlo radiation transport calculations prepared for the pre-commissioning Station Safety Report.

## External radiations

In most areas of containment, the external radiation field is dominated by intermediate and fast fission neutrons and by high-energy gamma rays from the decay of water activation products (e.g. $^{16}N$; gamma ray emissions at 6.4 and 7.1 MeV). However, the presence of activation and fission products, deposited as crud on the internal surfaces of pipes and vessels or present as solutes and colloids in the process fluids, still dominate the radiation fields around certain plant components.

Figure 1 shows the variation in doserates and the variation in neutron radiation quality throughout the reactor building annulus whilst at 100% power. The highest doserates are found on the upper levels of the building, especially in areas with line-of-sight to the Refuelling Cavity and RPV Head. Neutron quality is given by the k-factor (a higher k-factor indicating a harder neutron spectrum).

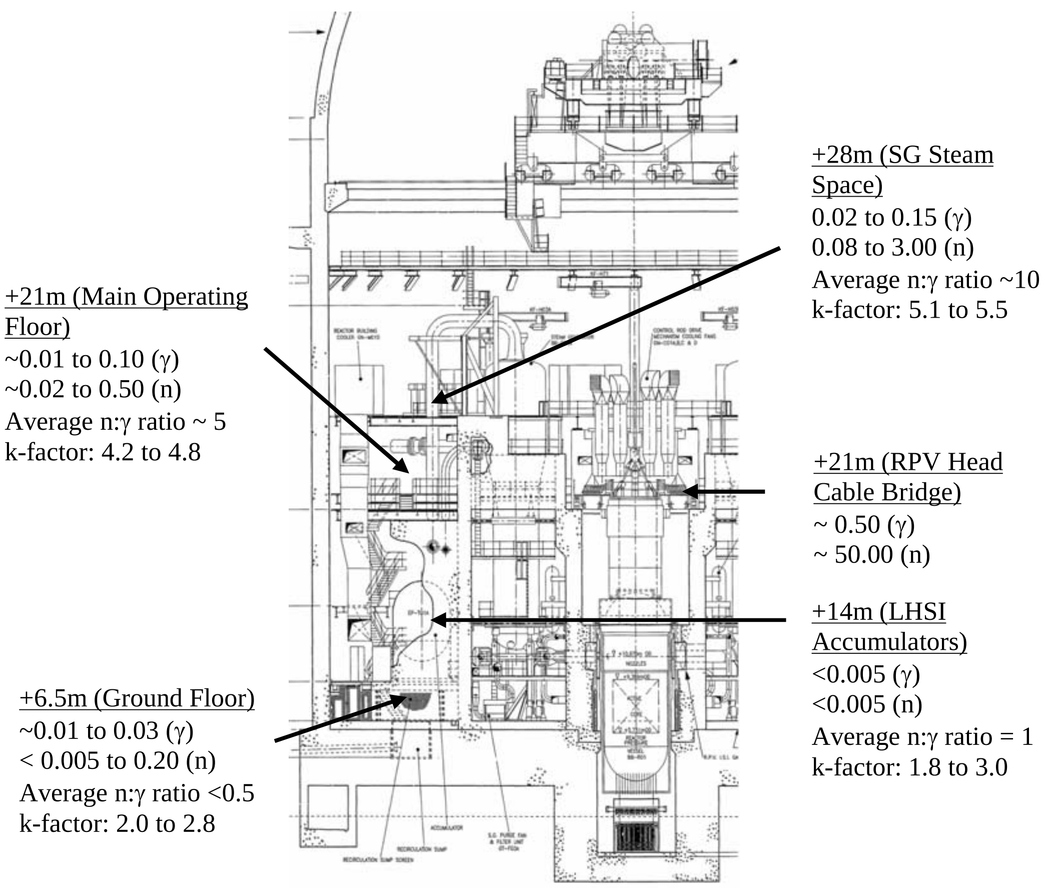
A negligible contribution to the external radiation field also comes from noble gases in the containment atmosphere (typically <100 Bq/m$^3$).

## Internal radiations

Low levels of activation and fission products (4 to 40 Bq/cm$^2$) are present as both fixed and non-fixed surface contamination inside containment.

Airborne radioactivity levels are usually low (<0.001 Bq/m$^3$ alpha, < 0.1 Bq/m$^3$ particulate beta and radioiodine). However, elevated levels of tritiated water (HTO) vapour, between 10 to 60 kBq/m$^3$, are found inside containment, giving an effective dose rate of approximately 0.2 to 1μSv/h. It is postulated that the source of this HTO vapour is gradual desorption of tritium from concrete and metalwork contaminated by a primary coolant leak during Cycle 5.

**Figure 1. Variation in radiation doserates (in mSv/h) inside the Containment Building, whilst at 100% power**



### *Justification of work at power*

Establishing a routine containment entry programme presents the opportunity for cost savings by reducing the scope of the Refuelling Outage. However, due to operational and nuclear safety restrictions on plant isolations at power, the tasks that can be performed are unlikely to be "critical path" activities; therefore one cannot make a robust justification argument based solely on critical path reduction. Other factors need to be considered, and a variety of arguments were used, either individually or in conjunction, to justify the decision to work at power. The principal arguments were:-

*Triviality of dose* – Although the whole of containment is designated as a High Radiation Area; many places in the annulus have doserates sufficiently low to make the conditions similar to rooms within the Auxiliary, Fuel and Radwaste Buildings, where no special access arrangements are necessary. As such, short duration jobs would accrue minimal dose, and a collective dose of less than 0.05 man.mSv was deemed to be trivial, requiring no further justification or optimisation.

*Lower doserates* – Some areas of containment have lower doserates at full power than when the unit is shutdown. This is principally due to different plant configurations, especially around the Residual Heat Removal System. In many other areas, doserates at power are not significantly higher than at shutdown.

*Improved industrial safety* – Some tasks (especially scaffold construction) in areas that would be highly populated during the outage, could be performed at power, without risk to persons that would otherwise be in that area at shutdown.

*Resource minimisation* – The draft outage plans had a number of resource peaks where demand for manpower and service equipment (e.g. scaffolds) was greater than supply. Working at power would enable these resource peaks to be flattened.

*Improved outage mobilisation* – Pre-staging and installation of radiological protection equipment (such as temporary shielding), would enable faster access to plant areas and improved radiological control during the first few days of the outage.

*Prevents a reactor trip* – Work to prevent an imminent reactor trip, was justified as it would keep the unit on-load, thus avoiding the dose associated with a forced outage maintenance plan and the subsequent plant operations required to return the reactor to power.

## Optimisation of doses

### *Engineered controls*

Airborne radioactivity levels were minimised by running the Mini-purge extract system for 2 to 3 days prior to each containment entry, which enabled the containment atmosphere to be cleaned at a rate of 7 200 $m^3$/h. Access to very high doserate areas inside the Bioshield was restricted by simply locking doors. As the radiological conditions in the annular areas are relatively stable whilst at power;

signs and barriers were used to identify low doserate areas, hotspots and radiation beams in order to prevent inadvertent access to other areas that could not be locked off.

### *Pre-job briefings and setting to work*

All staff entering containment received a detailed brief. Where available, Health Physics Information Sheets were given to each work party. These showed a photograph of the item to be worked, a map of its location and details of the expected radiological conditions in the immediate area.

### *Radiological measurements at the workplace*

RP Technicians ran air samples and conducted detailed surveys during the planning stages of tasks, to determine whether the proposed work area was tenable. They only accompanied work groups when personnel were accessing areas where steep doserate gradients existed or where significant intrusive work on active systems was being performed (e.g. valve replacements).

Where work was determined to be of low radiological risk and experience showed that radiological conditions were stable, maintenance teams were able to rely on their own specially trained staff, that were able to perform simple self-monitoring for gamma radiation and surface contamination (known as Radworkers). This allowed the work party to confirm the validity of the measurements made some weeks previously by the RP Technician. Use of Radworkers also enabled us to minimise the collective dose by reducing the RP dose burden.

## Assessment of doses

### *External radiations*

The main dosimetric problems associated with containment entries at power are the assessment of neutron dose and the presence of high doserate radiation beams that may not interact with personal dosimeters.

All staff entering containment wore a passive neutron dosimeter. Sizewell B uses the CEGB Albedo, which uses two lithium fluoride TLDs to measure thermal and intermediate neutrons below 25 keV. To account for neutron energies greater than 25 keV, Albedos are assigned a correction (or "k" factor). Detailed neutron spectra surveys had been performed throughout containment at various reactor power levels. These surveys had identified a range of k-factors between 1.8 and 5.5, as shown in Figure 1. All neutron dosimeters were assessed using the maximum k-factor of 5.5.

Sizewell B's legal beta/gamma dosimeter is the Siemens Mk1 EPD. Staff entering the reactor building at power had EPD alarms set at 500 μSv/h and 100 μSv. The dose alarm is 50% lower than that normally used in other controlled areas on-site; this was done in order to compensate for the neutron component not measured by the Mk1 EPD. As a practical indication of total dose (in the absence of a direct reading electronic neutron/gamma dosimeter), staff were instructed to assume that their total dose was in fact *10 times* the EPD reading when working on the 21m level and above, and *twice* the EPD reading when working on the 14 m level and below.

Where highly localised beams were present, access to these areas was simply prohibited, rather than attempting to multi-badge individual workers.

### *Internal radiations*

Under the Ionising Radiations Regulations 1999 [1], components of dose less than 1mSv are deemed to be non-significant and as such, no formal assessment is required (provided that the sum of unassessed doses remains less than 1 mSv). Using air sample data and airlock entry records to measure area occupancy, estimates of dose were made and tracked on a spreadsheet to ensure that no individual received a significant internal dose; therefore no personal air sampling, *in vivo* or *ex vivo* bioassay programmes were required.

### *Results of dose assessment*

The dosimetric results of the Cycle 5 and Cycle 6 containment entry programme are shown in Tables 1 and 2. Between 2001 and 2003, the total station collective dose received during normal power operation has remained constant at approximately 54 man.mSv. In 2001, the proportion of this dose received performing containment entries at power was 44%. In 2002, this proportion fell to 36%, but had risen again in 2003 to 51%.

**Table 1. Estimated doses for containment work activities, excluding radiological protection, by year**

| Calendar Year | 2001 | 2002 | 2003 |
|---|---|---|---|
| Neutron Collective Dose (man.mSv) | 15.120 | 7.590 | 16.800 |
| Gamma Collective Dose (man.mSv) | 2.775 | 4.232 | 4.991 |
| Collective Dose (man.mSv) | 17.895 | 11.822 | 21.791 |
| Number of people | 43 | 129 | 119 |
| Average Individual Dose (mSv) | 0.416 | 0.091 | 0.183 |
| Maximum Individual Dose (mSv) | 1.595 | 0.803 | 1.052 |

Table 1 shows that over the period 2001 to 2003, collective doses have increased, although average and maximum individual doses have fallen. Also, it is interesting note that the contribution of the neutron component to collective dose is between 2 and 5 times the gamma component.

The data for radiological protection staff (shown in Table 2) is not as complete as the data for bulk work activities, as RP staff were instructed to use the standard EPD task code, which has made the subsequent differentiation of gamma dose received in containment at power from other RP activities difficult.

**Table 2. Estimated doses for radiological protection activities inside containment, by year**

| Calendar Year | 2001 | 2002 | 2003 |
|---|---|---|---|
| Neutron Collective Dose (man.mSv) | 5.610 | 2.600 | 3.800 |
| Gamma Collective Dose (man.mSv) | ~ 2.000 | ~ 2.000 | 1.608 |
| Collective Dose (man.mSv) | ~ 7.610 | ~ 4.600 | 5.408 |
| Number of people | 8 | 22 | 22 |
| Average Individual Dose (mSv) | ~ 0.951 | ~ 0.209 | 0.246 |
| Maximum Individual Dose (mSv) | 1.590 | 0.430 | 0.781 |

However, this data shows that as the amount of work performed in containment grew, the numbers of RP staff required to manage these activities also increased. Twenty-two RP technicians and engineers were involved in 2002 and 2003 compared to just 8 in 2001. Over this period, the RP collective dose and the maximum individual dose fell, although the average individual dose rose to just under 0.25 mSv. Unlike the bulk of the containment work, the difference between the contributions of neutron and gamma radiations is less than a factor of 2.

The impact of using Radworkers is clearly demonstrated by comparing the 2001 and 2003 collective doses. RP dose contributed approximately 30% to the overall collective dose received in containment at power in 2001, when use of Radworkers was minimal. In 2003, the scope of work enabled much greater utilisation of Radworkers and as a result, the RP contribution to containment collective dose fell to 20%.

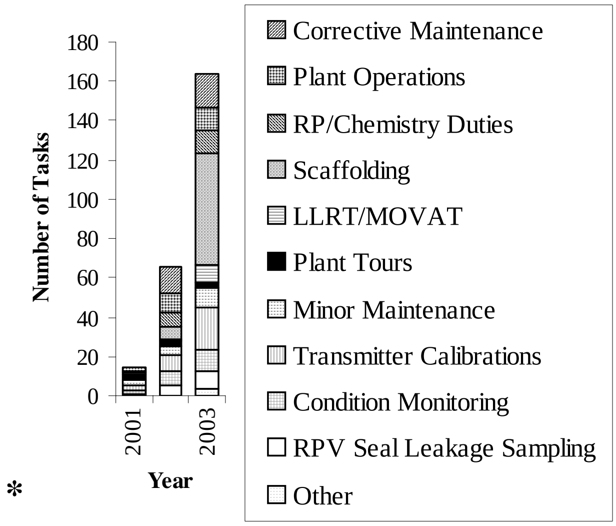
## Scope of work performed

The definition of "task" used in Figure 2 is simply a single Work Order. A better measure would be number of man-hours in each task category. Unfortunately, it was very difficult to obtain an accurate estimate of this parameter from our work management computer system, and information for 2001 is extremely unreliable.

Despite these limitations, a clear trend shown in Figure 2 has been the increase in the amount of work performed in containment at power since 2001. This has increased from approximately 30 tasks in 2001 to over 160 in 2003. In addition, the data is sufficiently robust to show that the relative contribution of each type of task has varied considerably over the period shown. Most tasks performed in 2001 were plant tours required to identify the location of a primary coolant leak (which caused a forced outage in March 2001) and the intensive leak searches and corrosion monitoring surveys subsequently required as part of the return to power safety case.

In 2002 and 2003, regular entries were made for 1 to 2 days per month, increasing to 6 days per week in the month prior to the refuelling outages (RF05, May 2002 and RF06, October 2003). Mandatory leak searches and corrosion surveys were still being performed every 6 to 8 weeks, but additional tasks were incorporated to maximise the cost-effectiveness of the containment entry. Tasks included scaffolding, transmitter calibrations and plant operations. Corrective maintenance was also performed, mainly to keep a defective Emergency Boration System valve actuator operable. Significant modifications to the Steam Generator ventilation ductwork were also performed during the pre-RF06 period.

**Figure 2. Number and type of tasks performed during containment entries, by year.**
*** denotes data for this period is significantly underestimated (see text for further detail)**




## Discussion

International experience with containment entries at power varies considerably. Most European utilities only undertake containment entries to rectify faults that threaten an imminent reactor trip, where as some North American utilities have established a routine containment entry programme. For example, Three Mile Island performs entries every 6 to 8 weeks to execute a similar range of tasks as Sizewell B. The programme at TMI accrued 12 man.mSv, equivalent to 20% of their normal operation dose in 2002 [2].

Whilst planning these entries, there was little published data or operational experience available to identify a suitable dose constraint. The individual doses received during this programme were low compared to national dose limits and a company dose constraint of 10 mSv [3]. Collective doses were also low, although relative contribution to overall normal operation dose was approximately twice that at Three Mile Island. In terms of dose, the most significant tasks in 2001 and early 2002, were the primary coolant leak searches and subsequent corrosion monitoring inspections. By 2003, the most radiologically significant tasks were scaffold construction on the upper floors of containment. Doses to radiological protection staff were mostly received installing temporary shielding (on the lower floors) and accompanying System Engineers on leak searches etc. This difference in work area explains the variation in the neutron:gamma ratios between RP tasks and maintenance tasks highlighted in the results section.

The results show that the neutron component of dose dominates when working inside containment. This is partly due to the high neutron:gamma ratio found on the upper floors, the conservative choice of k-factor and also the wide variation in limits of detection for the albedo (~50μSv) and the EPD (<1μSv). Prior knowledge of the neutron spectra is essential to avoid significantly underestimating dose. For example, workers at another UK power station received approximately 11 man.mSv (3.7 mSv, maximum individual dose) when working close to a Bioshield penetration. Lack of knowledge of the neutron spectra had led RP engineers to underestimate neutron doserates. This lack of recognition persisted, even after having confiscated the workers' neutron-activated jewellery and clothing at the RCA exit monitor [4].

As part of the ALARA review, it is important to establish the "usefulness" of the work performed. A quantifiable benefit was a reduction of 3 days in the RF06 critical path, by modifying the Steam Generator ventilation ductwork whilst at power (for a collective dose of approximately 1.5 man.mSv). Less discernible benefits were derived from other tasks. Individually, small tasks such as transmitter calibrations have negligible impact on outage scope and dose. And when many tens are performed together, the contribution to outage workload reduction is still rather small, but the radiological impact with respect to normal operation dose can become significant.

## Conclusions

This work has shown that a wide range of tasks can be performed inside a containment building at power, for comparatively low individual and collective doses (although these represent significant proportions of the normal operation dose). However, to achieve these outcomes, an extensive input from RP engineers and technicians was required. For certain tasks, such as scaffolding and lagging on the RHR system, doses are clearly optimised by working in containment at power. However, the doses received on some other tasks, may not have been ALARA, especially during 2003. This paper recommends that further refinement of the justification arguments is necessary and that annual dose constraints of 1.5 mSv and 15 man.mSv are implemented for routine containment entry programmes at Sizewell B.

## References


1. *Work with ionising radiation*. Ionising Radiations Regulations 1999. Approved Code of Practice and Guidance L121, HSE Books, 2000, ISBN 0 7176 1746 7, Paragraphs 385 and 386.

2. Velez, P., *Three Mile Island*, Amergen Energy, Personal Communication.

3. Morris, S., *CDRL – Company Dose Restriction Level. Occupational Exposure Management at Nuclear Power Plants, Proceedings of the 3rd ISOE European Workshop*, 2002. pp. 215-223, ISBN 92 64 02135 3.

4. EAR PAR 03-046. www.wano.org 27th August 2003.

# A CORPORATE ALARA ENGINEERING SUPPORT FOR ALL EDF SITES
# A MAJOR IMPROVEMENT: THE GENERIC WORK AREAS OPTIMISATION STUDIES

**A. Quiot *, J. Lebeau ****
*EDF/UTO – National Radiation Protection Engineering, **EDF/ALARA Project, France

**Summary**

ALARA studies performed by EDF plants are quite simple and empirical. Most often, feedback experience and common sense, with the help of simple calculations allow reaching useful and efficient decisions. This is particularly the case when the exposure situations are not complex, within a simple environment and with a single source, or one major source. However, in more complex cases this is not enough to guarantee that actual ALARA solutions are implemented. EDF has then decided to use its national corporate engineering as a support for its sites. That engineering support is in charge of using very efficient tools such as PANTHER-RP. The objective of the presentation is to describe the engineering process and tools now available at EDF, to illustrate them with a few case studies and to describe the goals and procedures set up by EDF.

## A strong EDF commitment to facilitating ALARA implementation on sites: the corporate engineering support

### *Initial situation*

ALARA studies performed by EDF plants are quite simple and empirical. Most often, feedback experience and common sense, with the help of simple calculations allow reaching useful and efficient decisions. This is particularly the case when the exposure situations are not complex, within a simple environment and with a single source, or one major source.

However, in some cases this is not enough to guarantee that actual ALARA solutions are implemented. Common sense is not able to handle complex situations when many sources contribute to the dose rate at the workplace, when the workloads at the same workplace are very different from one outage to the other and when some old materials may be removed or new ones installed during an operation, modifying then the radiological context. Furthermore the single use of feedback experience may lead to maintain practices without taking care of progresses due to technological and knowledge improvements. Therefore, it is then necessary to perform more complex analysis relying on the use of quite sophisticated radiological protection software’s and codes. Such codes are not available at the site level, where there are no resources (specialists and time) to use them.

## Objectives and resources

Since 1991, EDF has established a national ALARA programme with a very effective result in terms of dose reduction. Of course, EDF management has decided to further improve occupational exposure management and dose reduction (both collective and individuals). One key element allowing reaching the new goals is the set up of a national corporate engineering as a support for EDF sites for quite usual interventions.

That engineering support consists of a growing up team comprising at the moment about ten engineers, including CAD specialists and health physicists. It is in charge of using very efficient tools such as PANTHER-RP to perform national modelling studies concerning the reactor and auxiliary buildings areas, which are the most costly in terms of doses. That tool has been developed initially for the first steam generator replacements by EDF SEPTEN engineering department. It uses up to date and friendly user 3D software's to create a geometrical model of the concerned area with all existing materials (pipes, valves, concrete walls…) allowing visualising on personal computers, each area from all perspectives.

Other important inputs for PANTHER RP are the quantities of radioisotopes present in each material. The code allows then estimating the dose rates at each location in the area, calculating the contribution of each equipment (i.e. sources) in the area to the dose rate in each point; calculating also the contribution of each radio element to the dose rates.

With the help of these models, the engineering is then able to perform in depth generic work areas optimisation studies, taking into account the workload in each workstation. Up to recently these studies were performed only for huge operations such as steam generator replacements, they are now proposed to EDF sites for more usual interventions. The selection of these interventions takes care of – the dosimetric cost of the operation(s) performed at the workstation(s); – the complexity of the environment (multiple sources); – the repetitiveness of the jobs (either on a single unit or on several ones). One may estimate that there are about ten such situations per type of reactor: operations performed in the vicinity of the reactor coolant valves, operations performed in the reactor pool, operations performed in the vicinity of the secondary side of the steam generators, maintenance interventions on the DHRS and CVCS heat exchangers…

## The generic work stations optimisation studies: example of the work areas in the vicinity of the primary coolant valves.

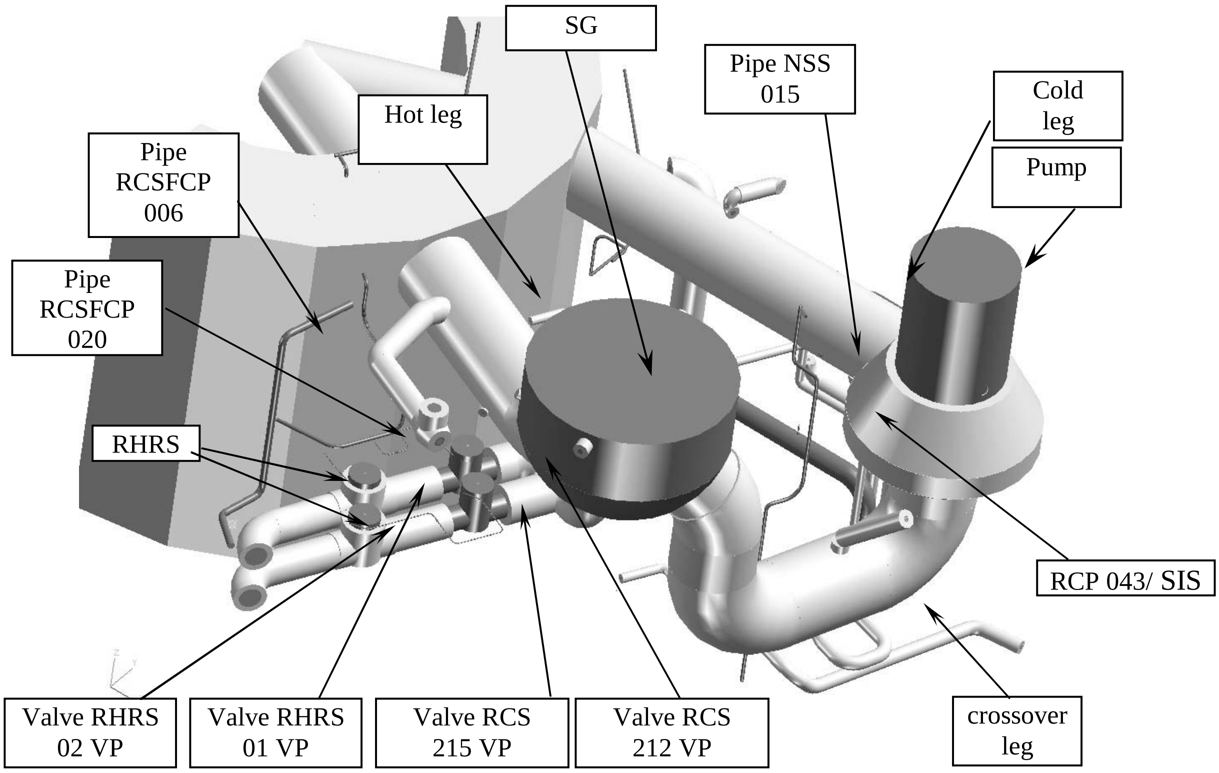
One example has been selected here. It concerns the possible reduction of occupational exposure at the workstations situated nearby the primary coolant valves (between 30 and 150 man.mSv before optimisation according to the contamination level of the circuits for the 900 MW units).

As regards the three criteria already mentioned:

- The dosimetric stake is important not only in terms of collective dose but also for individual doses, as only 15 workers are concerned.

- Secondly, more than ten materials (or parts of) are contributing-sources to the dose rates at the workstation (primary pipe hot leg, primary pipe cold leg, U primary pipe, RCS/DHRS valves, Reactor cavity and spent fuel pit cooling and treatment system pipes…) as may be seen on the Figure one, which is the result of the geometric analysis with the 3D software.
- Finally, at each outage, some inspections are performed on these valves.

**Figure 1. Location of sources in the surrounding area**




The number and type of inspections depends on the type of outage. Three main work scenarios are possible:

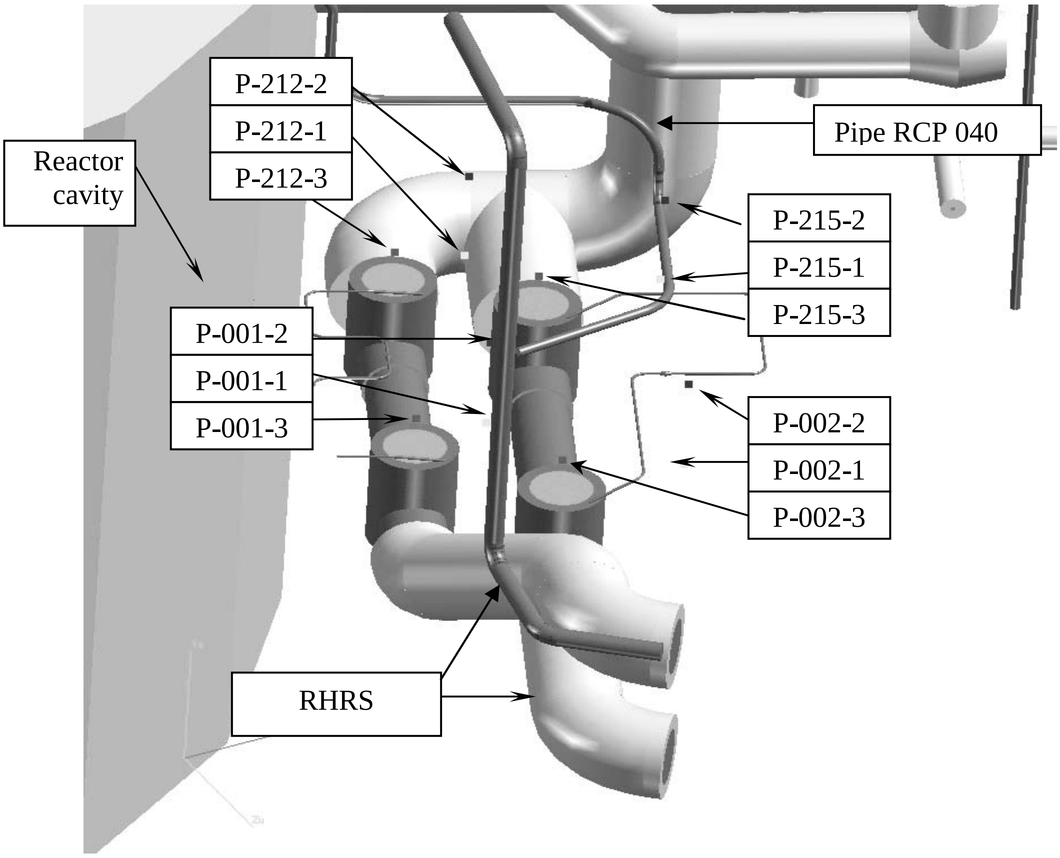
- Scenario 1: simplified inspection of one Reactor Coolant System/Decay Heat Removal System valve.
- Scenario 2: one complete inspection and one simplified inspection of one Reactor Coolant System/Decay Heat Removal System valve.
- Scenario 3: one complete inspection of two Reactor Coolant System/Decay Heat Removal System valves and one simplified inspection of one Reactor Coolant System/Decay Heat Removal System valve.

The possible options are the installation of biological shielding (different thickness), optimisation of water movements, chemical decontamination of the RCS and DHRS valves and nearby pipes with different processes, flushing, and removal of active materials.

The study is first performed using radiological data such as the contact dose rates and the sources spectrum for each material (or part of) from a representative unit (here Tricastin 1) with no specific pollution or hot spot.

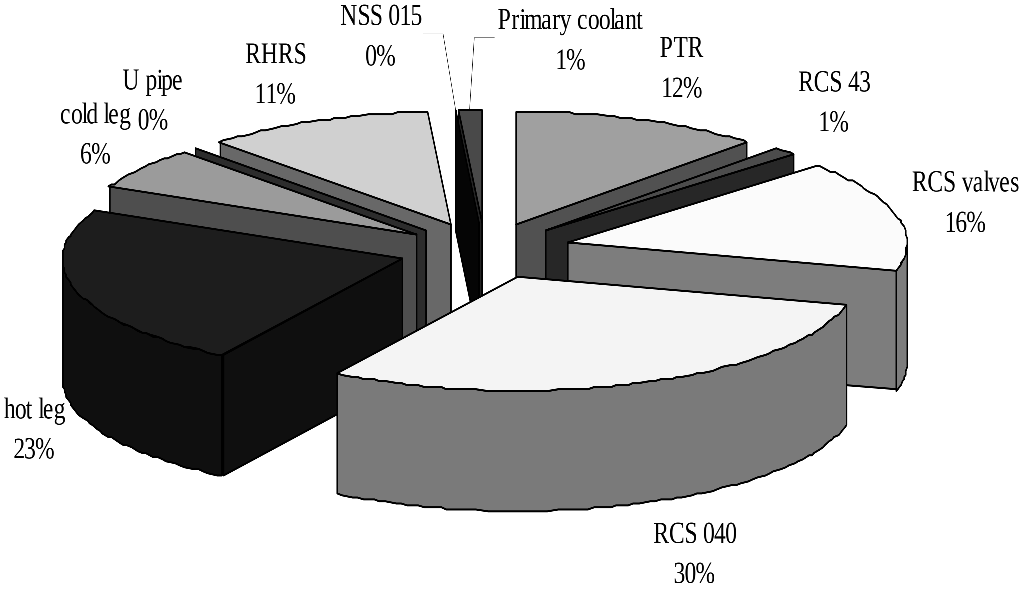
It is then possible to locate all precise positions of the workstations with regards to the different materials as may be seen on Figure 2.

**Figure 2. Location of the workstations**



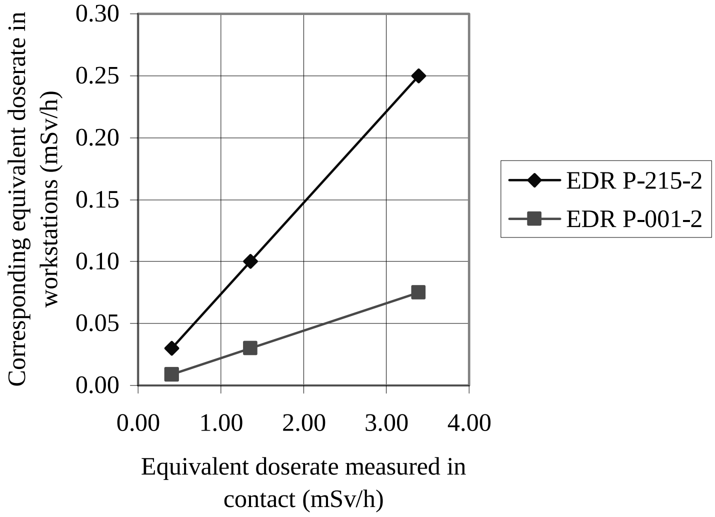

Knowing the location of the workstation and the radiological environment data, PANTHERE-RP allows providing the contribution to the dose rate at each workstation from each source, as illustrated for the workstation RCP 215 VP in Figure 3.

**Figure 3. Contribution to the dose rate at the workstation RCP 215 VP from each source**




It is then possible to provide an abacus providing the relationship between a contact dose rate (for a specific material) and its contribution in terms of dose rate at a workstation. This is illustrated on Figure 4 for the impact of the source “pipe RCP040” on the dose rate at the workstation RCP 215 VP. Taking into account the time spent at the workstation, and the measured dose rate on the pipe RCP 040 in a specific plant, the dose due to that specific material (i.e. source) is now easily estimated at the workstation RCP 215 VP. It is also therefore quite simple for that Plant to test the efficiency in terms of dose reduction of protection actions such as biological shielding installation between the source and the workstation. EDF has set up a pragmatic decision making rule: the installation of a biological shielding is worthwhile any time the dosimetric cost of its installation is overcompensated by more than 20%. In the case of the pipe RCP 040, it has been envisaged to install 1 500 x 300 mm lead blankets (6mm thick) in two thickness at the workstation side (option 2.1), or the same in two layers, i.e. four thickness at the workstation side (option 2.2).

**Figure 4.**




The combination of the three above mentioned scenarios with the two options and the decision-making rule lead to the following selection of option as a function of the contact dose rate on the pipe RCP 040.

**Table 1. Selection of biological shielding option at the workstation RCP 215 side as a function of the contact dose rate on RCP 040**

<table>
<tr><th></th><th>0.2 mSv/h ≤ d < 1 mSv/h</th><th>1 mSv/h ≤ d < 2 mSv/h</th><th>2 mSv/h ≤ d</th></tr>
<tr><td>Scenario 1</td><td colspan="3">Option 2.1</td></tr>
<tr><td>Scenario 2</td><td colspan="2">Option 2.1</td><td>Option 2.2</td></tr>
<tr><td>Scenario 3</td><td>Option 2.1</td><td colspan="2">Option 2.2</td></tr>
</table>

The previous table shows that a simple decision tool is now provided to all 900 MW units as an output from the use of more sophisticated tools by the national engineering support.

Another output from PANTHERE-RP is to providing the influence of each radio element, from all sources (and each source) on the dose rate at any workstation. This is of particular interest when analysing the efficiency of decontamination. In the previous example, it is foreseeable to decontaminate part of the circuit as shown on Figure 5 with the chemical EMMAC process.

**Figure 5. Part of the circuit to be decontaminated**

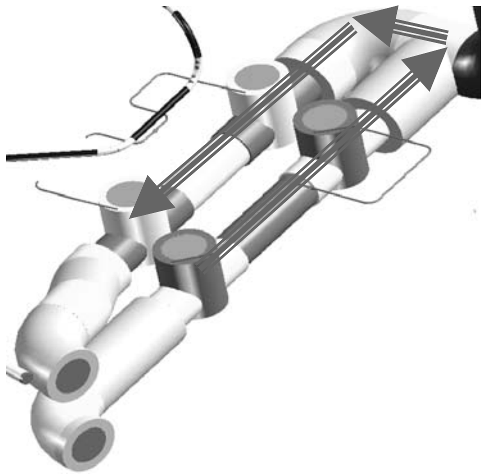
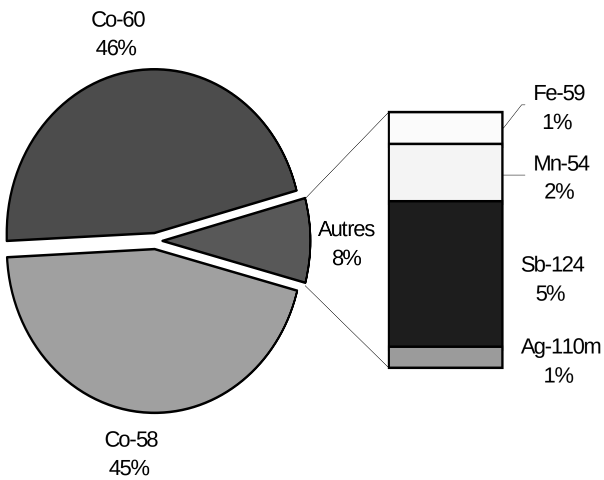


**Figure 6. Contribution of the different radio elements to the doserate at the workstation**




On Figure 6, one may notice that Cobalt 58 and 60 generate nearly all (92%) the dose rate at the workstation, in the same proportion (about 46% each). Having in mind that recontamination from $Co_{58}$ will reach the same level after only one operating cycle, while the level from $Co_{60}$ will only be reached after seven cycles, it may be estimated that, even if the decontamination factor is 100%, at the end of the next cycle already 46% of the previous contamination will be present again in the circuit due to Cobalt 58.

When decontamination is implemented, it will be performed after the installation of shielding and opening of the valves, but before the maintenance itself. The software allows then estimating a reduction by 45% of the collective dose for the jobs during that outage. The later years the decontamination will impact all the doses (including installation of shielding) but with a reduced efficiency. Therefore the percentage of reduction of collective dose after one year will only be 22.5% and after two years 20%.

The decision to do or not to do decontamination is not as trivial as the one for installing the shielding as the estimated cost of the decontamination is 100 k€. In that case a reference monetary value of the avoided man milli Sievert of 1 800 € is used. A cost benefit analysis is performed. It shows the importance of the sensitivity analysis, particularly when the ratio Cobalt 58/Cobalt 60 varies, which is the case when going from one plant to another.

It is then very important for each unit to be able to have quite a good knowledge of its sources spectrum. This will be easier to do with the new portable spectrometer in test at EDF. It is then interesting to note that the tools developed for modelling are totally complementary to those developed for measuring dosimetric data.

**Relationships between the site and the national engineering support**

As illustrated in the previous example, the studies performed by the engineering support will allow:

- Determining the contribution of each source and radio element on each dose rate at a workstation.
- Characterising exposure situations by answering to where?, when?, and how are doses undertaken?
- Identifying radiological protection options for reducing the exposures.
- Quantifying the efficiency of these options.
- Selecting the most pertinent options within an associated validity domain.

Of course this will only be achieved with a good co-operation between the plant teams and the national support. Most often, a plant originates the demand of a study after a first analysis by a local multi-disciplinary team, including health physicists, technical specialists and when necessary operators, planner. After a study of the demand by the national support, a kick off meeting between the plant and the national engineering support allows discussing and freezing the maintenance scenarios presented by the plant, checking the exposed workload data provided by the plant for each scenario, defining precisely the position of each workstation, and discussing all available feedback experience information both on dose-rates and radiological protection actions performed in the same area at different occasions (in the plant or in other similar plants).

As of necessity, the national corporate engineering makes then complementary spectrometry measurements to have a better knowledge of the spectrum of each source.

All the data being made available the national engineering support proceeds to the study and issues several documents (radiological context presentation document, geometrical context presentation document by the CAD team, document on optimisation of radiological protection for each maintenance scenario, validation feedback document after implementation…).

In the optimisation of radiological protection document the French sites are provided, as often as possible, with abacuses or synthetic tables allowing them to select the most adapted solution corresponding to their own situation.

**Conclusion 2004-2007**

During the next four years, most of the models corresponding to all interesting situations will be created both for 900 MW units and 1 300 MW units. It is a very important industrial investment but there will be soon a return on investment by allowing quick answers to questions from the sites. In fact, the national support studies may take a few man-hours to a few man-months depending on the complexity of the situation. The modelling of more and more areas in the reactor building being available, the new demands from the plants will be quicker to be answered to.

# ADVANTAGES OF COMBINING GAMMA SCANNING TECHNIQUES AND 3D DOSE SIMULATION IN DOSE OPTIMISATION PROBLEMS.

**F. Vermeersch**
SCK•CEN, Belgium

## Abstract

In this paper we present a method of combining results from gamma scanning equipment with a 3D dose simulation tool with the aim to achieve a reliable dose characterisation of the work site in order to perform dose assessment and optimisation for work planned in the area.

A first step in any ALARA pre-job study is the radiological characterisation of the work site. Traditionally this is done based on 4π dose measurement and spectral analysis of sweeps or samples taken from the site. This method can be very tedious and dose intensive especially in complex geometries. In recent years equipment such as gamma cameras and gamma scanners started to appear on the market enabling a remote localisation of source positions and geometry. We will show how the data of the gamma scanning can be analysed with a 3D dose assessment tool in order to achieve a reliable source model for the work environment and how the source and geometry modelling information can be used in the dose optimisation problems.

## Introduction

Planning activities in an irradiating environment involves the technical analysis of the work but also the assessment and optimisation of the occupational dose in order to comply with the ALARA requirements.

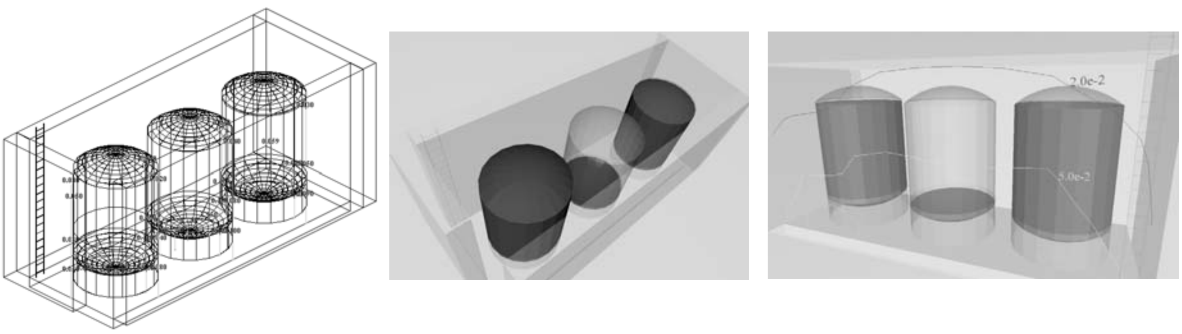
This is a complex task requiring the treatment of data going from source strengths, shielding, site geometry, work duration to even the distribution of the work force. Optimising the dose also means that different work scenarios should be compared to select the one being a good compromise between effort (financial, technical...) and dose reduction. Therefore, a need exists for a tool to simulate the different planned activities in order to evaluate the dose prior to the operation. In order to do so SCK•CEN developed the VISIPLAN 3D ALARA planning tool to assist the ALARA analyst in the field of dose assessment and optimisation [1]. The tool allows making a dose assessment in a 3D environment based on a point-kernel calculation corrected with an infinite media build-up factor. VISIPLAN has in the past years proven to be a valuable tool for the ALARA analyst [2-6].

The aim in the pre-job study is to establish an adequate radio-geometrical model of the site enabling a good dose calculation for the work. With adequate we mean a model with a level of detail suited for both calculation speed and required accuracy for the dose assessment in the field of radiation protection.

However before any calculations can start we need to gather information on the geometry, materials and sources present on the site. A major part of the geometry and material information can be found in the technical descriptions and plans of a site. In some cases there exists the need to re-measure the positions and dimension of some infrastructures because they where not build according to plan or they were adapted during the lifetime of the site. In those cases we can resort to techniques like laser scanning to establish relatively quickly an as build plan in a 3D CAD format.

The radiological characterisation of a site is more difficult to achieve. Traditionally this is done using a set of $4\pi$ dose measurement at different positions of the site together with spectroscopic analysis of sweeps or samples taken from the sources. This method can be very tedious and dose consuming for complex industrial environments, especially if little information is available on the geometric extend and exact position of the sources. The dose rate map established by direct measurement can be used to assess the dose, under the condition that the radiation field does not change during the operation as a consequence of geometry changes or source removals. When we want to predict doses in changing work environments we need to establish the information on source location, source strength and source composition. Sometimes it is possible to derive the source position, composition and geometry from the analysis of the technical data of the plant. Source strengths can then be derived by fitting the calculated dose rates to the measured dose rates; a technique applied in the source fitting routine available in VISIPLAN (Figure 1). This is a practical method but can in some cases lead to missing the contribution of some hot spots that where difficult to measure due to geometric restrictions (difficult to access with the dose measurement device).

**Figure 1. Example of a source strength assessment based on the dose measurements distributed over the site (a). The positions of the main sources (in red) are derived from the technical data of the site (b). The source strengths are determined by fitting calculated dose rates to the measured dose rates and can then be used to determine the dose rates at different positions in the work area (c) (dose rates expressed in mSv/h).**




(a) (b) (c)

In recent years however equipments like gamma camera's and gamma scanners started appearing on the market enabling an easier, remote localisation of sources or hot spot on a site. In this paper we show how these devices can help in the characterisation of a site and can help to establish an adequate radio-geometrical model of the work place. This is demonstrated on an application in an industrial environment.

Part of the work presented here was performed as part of the VRIMOR European 5$^{th}$ framework programme on "Virtual Reality for Inspection, Maintenance, Operation and Repair" where the viability of the integration of different technologies like gamma scanning, geometrical scanning, radio-geometrical modelling and human motion simulation were explored [7-8]. The results presented here concentrate on the radiological modelling aspects of the work, especially on the interpretation of gamma scanning results.

First we will give a short description of the gamma scanning equipment used and introduce the method we developed to analyse the gamma scan with VISIPLAN. Finally we demonstrate the method in the characterisation of an industrial environment at a nuclear power plant.

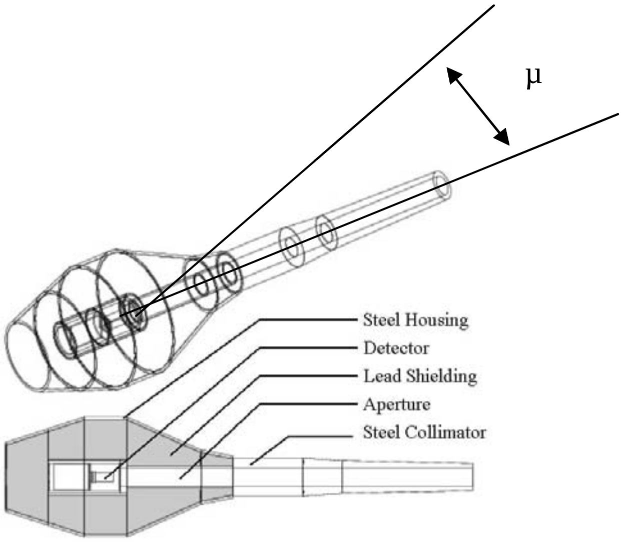
## Gamma scanning and gamma scan interpretation

The method to interpret gamma scans is based on the application of the EDR-scanner develop by CIEMAT (Spain) [7-8]. The scanner integrates three sensors, a collimated gamma detector a video camera and a laser distance meter. The gamma detector is a Cs(Tl) crystal coupled to a photodiode with an energy threshold in the 150-200 keV range. The detector is located in a stainless steel housing with a lead shielding as can be seen in Figure 2. The effective shielding is about 5 cm lead with a higher shielding value in the area surrounding the collimator opening. The collimator aperture used for the measurement is ±4°. The whole system is mounted on a pan and tilt platform enabling an automatic scan of the area. Spectra are measured in the different detection directions and stored in a 25 energy bin format together with the collimator direction and the distance to the measured object. A special interface was developed to transfer and display the measured results in VISIPLAN.

The interpretation of the gamma scans involves two parts. A first part concentrates on the visual interpretation of the scans, overlay images are used in order to determine the position of hotspots or the geometry of the sources. It is recommended that a series of scans are taken from different positions on the site. This enables to determine source positions using triangulation and reduces the risk of associating a source to the wrong object. This analysis leads to a first suggestion for the source distribution of the site. The model is then confronted with the available technical data of the site in order to qualitatively check that the proposed source distribution is a good candidate.

The spectroscopic capabilities of the scanner are used to determine the isotope vector important for the dose assessment. The isotope vector data can be further enriched by introducing data obtained through spectroscopy on samples taken from the content of certain volumes.

**Figure 2. Layout of the EDR gamma scanner (CIEMAT)**
μ indicates the angle enclosed by the collimator direction and a point source direction




Once the source geometry is established we can perform a quantitative analysis of the scans. This means that we will try to determine the source strengths on the site by comparing calculated scan results with the measured ones.

In order to do this we need to establish a relationship between the effective dose at the detector position and the response of the detector to this dose. This can be done by determining the response of the scanner for different detector orientations to the radiation field generated by a point source emitting photons at energy En. This will establish a relationship between the Instrument dose rate "IDR" and the effective dose rate at the instruments position for a gamma source emitting at energy En. Taking into account the axial geometry of the scanner we can define the instrument response function depending on then energy En and on the angle μ (Figure 2). The relationship between the instrument and the effective dose rate is given as:

$$\varepsilon(En,\mu).E(En) = IDR$$

and

$$IDR = \sum_i h_i.CPS_i$$

with $h_i$ the dose conversion factors for a rotational irradiation geometry and $CPS_i$ the counts per second detected in the energy bin $En_i$. The directional sensitivity of the detector-collimator couple was measured for a $^{60}$Co and a $^{137}$Cs source at a distance large enough to generate a plan-parallel radiation field at the detector position. The derived dose rate response function is the basis for the source strength evaluation method used in VISIPLAN-VISIGAMMA.

The source strengths are determined by fitting the simulated gamma scan expressed in IDR-values to the measured one. The simulated scan is calculated taking into account the geometry, material and source information in the model and the energy dependent directional dose response function of the EDR scanner.

This methodology was first tested in the laboratory, in a controlled environment consisting of two well known sources. The method performed well and was able to determine the source strengths within 20 and 30%.

## Demonstration of the method in an industrial environment

A demonstration of the method was performed within the VRIMOR project. The industrial area selected is part of the auxiliary building of Almaraz Nuclear Power Plant (Spain) [8].

A geometric scan of the area was performed by Z+F Ltd using their "Imager 5003" laser scanner. A CAD model was created based on the measurements and then transferred to a VISIPLAN model including material information. The materials data associated to the volumes were gathered on-site by Tecnatom.

The geometric scan was followed by the gamma scanning campaign performed by the CIEMAT team. The distance and orientation data of the EDR-scanner are fitted to data of the geometry scan in order to determine the EDR position in the CAD, respectively the VISIPLAN model.

The results of the geometric scan and the model derived from it in VISIPLAN are given in Figure 3.

Two gamma scans were used in the radiological characterisation of the site, their positions are also shown in Figure 3. The overlay images of the scans are given in Figure 4.

**Figure 3. VISIPLAN model of the site geometry indicating the position of the gamma scanner during the two scans**

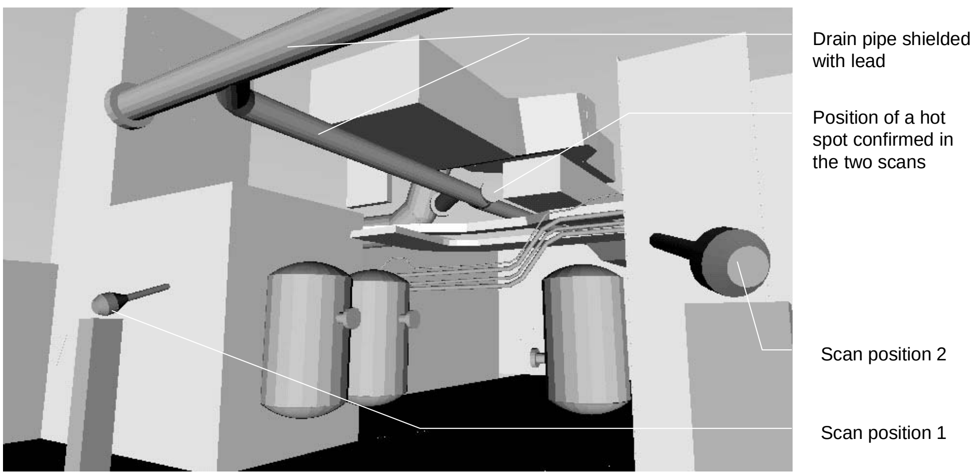
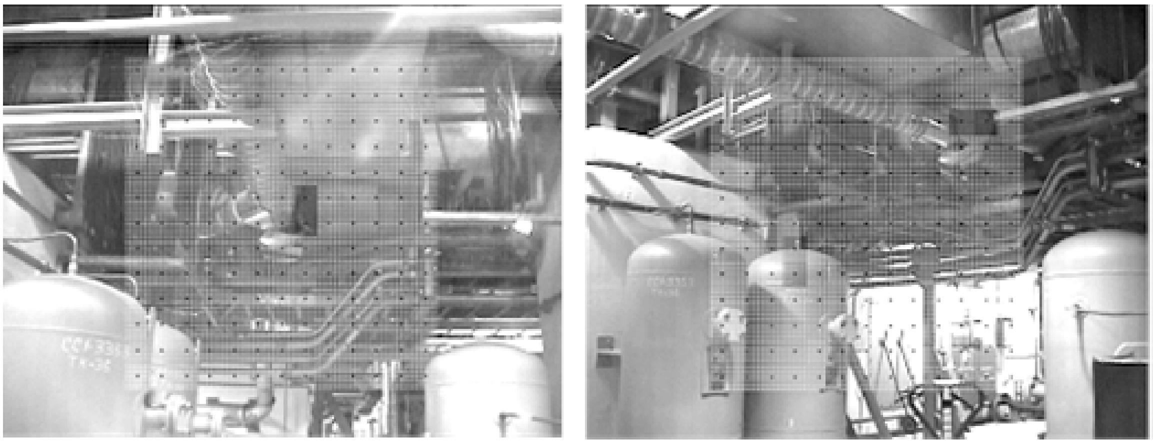


**Figure 4. Gamma scan intensity overlay image taken from two scanning positions (dark grey indicates higher gamma intensity)**



A hot spot can be seen at the tube with the end flange. The position of the hot spot is confirmed in the second scan taken from another position. The spectral analysis of the measurements suggests that $^{60}$Co is the pre-dominant isotope, so it was decided to continue the analysis with $^{60}$Co equivalent sources.

A first attempt to simulate the scans using one source, positioned at the hot spot, failed and leads us to further analyse the technical data of the site. The technical information gathered by Tecnatom suggests simulating the area using the source distribution presented in Figure 5. Three cylindrical volumes are used representing the source A, B and C in the tubes. This model proved to be more realistic and could account for the high background detected in the gamma scanner signal. Based on this model we determined the source strength of the A, B and C sources (Figure 5). A good agreement was now found between the simulated and the measured gamma scans.

Once the source strengths determined we calculated the dose in the area with the VISIPLAN tool and compared the calculated dose rates with the dose rates measured on site (Figure 6). An agreement was found within 20-30%, a good agreement considering the accuracy of the point-kernel calculation method used in VISIPLAN and the gamma scan calibration method proposed for the gamma scan interpretation.

It is interesting to notice that the direct viewing of the scans would lead us to believe that only one important source (source A) is present in the scene. However the detailed analysis using the 3D model in VISIPLAN showed that the drain pipes B and C are also major contributors to the dose. Source B lays only partly in the field of view of the scans and source C is outside the field of view but they account for the high background detected in the scans. This analysis was only possible because we performed a thorough calibration of the gamma scanner in all directions and could account for the contribution of source B and C to the signal.

**Figure 5. Simplified model of the area including the source distribution derived from the analysis of the scans and the technical data of the site**

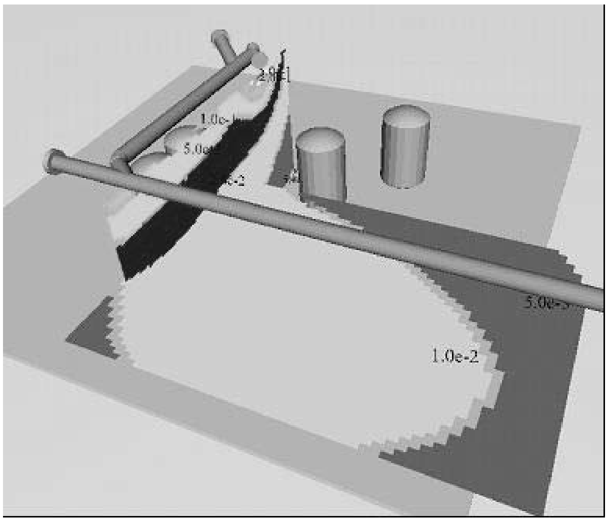
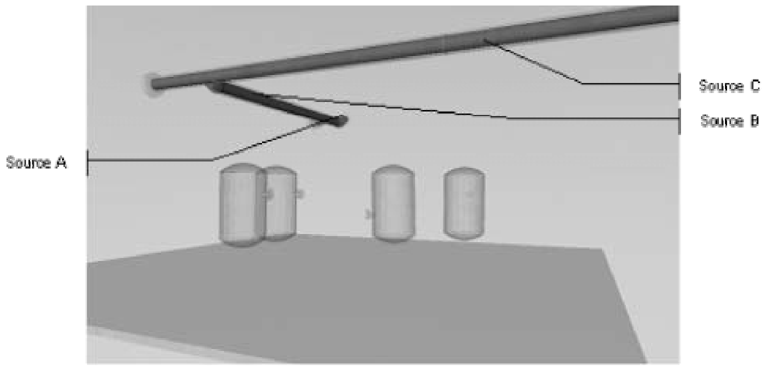



**Figure 6. Dose rate map calculated in two planes of the area using the VISIPLAN tool**




## Conclusion

The standard radiological characterisation of a site can now be augmented by using devices such as gamma scanning in order to determine source positions, source geometry and source composition. Hot spots that could be missed by traditional methods can now be picked up through the gamma scanning.

A thorough $4\pi$ calibration of the gamma scanner combined with the use of a radio-geometrical model makes it possible to perform a quantitative analysis of the source strengths leading to an adequate radio-geometrical model of the site that can be used in dose assessments and optimisation for work planned in irradiating environments.

*Acknowledgements*

Part of the work presented here was performed in the European 5th framework programme VRIMOR "Virtual Reality for Inspection, Maintenance, Operation and Repair", FIKS-CT-2000-00114. The partners in the project were NNC Ltd Manchester, (coordinator and ERGODOSE human modeling development), Tecnatom s.a., Madrid, (Evaluation and logistics with NPP site), SCK•CEN, Brussels (VISIPLAN-VISIGAMMA development), CIEMAT, Madrid, (Gamma scanner development), Z+F Ltd, Manchester (geometric scanning) and Universidad Politecnica Madrid (HeSPI human modeling development). Special thanks to Almaraz NPP for providing the industrial test site.

**References**

1. Vermeersch, F., C. Van Bosstraeten, *"Development of the VISIPLAN ALARA planning tool", Proceeding of the International Conference on Topical issues in Nuclear Radiation and Radioactive Waste Safety*, 31 August-4 September 1998, Vienna, Austria.
2. Vermeersch, F. and Van Bosstraeten, C., *"Software VISIPLAN: A powerful tool for optimisation*", 2nd European workshop on occupational exposure management at NPPS, 5-7 April 2000, Tarragona, Spain.
3. Massaut, V., P. Deboodt, F. Vermeersch, A. Lefébvre, M. Klein, *"Optimisation and decommissioning: Challenge and limits"*, 1st European ALARA Network Workshop, ALARA and Decommissioning, 1-3 December 1997, INSTN Saclay France, Paper No. 1.
4. Eggermont, G., N. Buls, A. Hermanne, W. Van Suetendael and F. Vermeersch (1997), "Optimisation opportunities in the decommissioning of accelerators", 1st European ALARA Network Workshop, ALARA and Decommissioning, 1-3 December 1997, Paper No. 14., INSTN, Saclay, France.
5. Massaut, V. (2002), "The radioprotection Optimisation applied in an actual case: the BR3 decomissioning project", *3rd European forum of "Radioprotectique*".
6. Antoine, Ph. 2002, *Démantèlemant de la cellule chaude n°41 au SCK•CEN en s'aidant de l'outil de planification ALARA VISIPLAN*, 3ème Journées SFRP, 11-12 juin 2002.
7. Lee, D.J. *et al* (2001), "Virtual Reality for Inspection, Maintenance, Operation and Repair of Nuclear Power Plant (VRIMOR)", FISA 2001, EU Research in Reactor Safety, 12-15 November 2001, Ed. by G. Van Goethem *et al.*, Luxembourg,
8. VRIMOR project: "*Technological and user Perspective report*", FIKS-CT-2000-00114, European 5th framework programme.

## CONCLUSIONS AND RECOMMENDATIONS FROM THE 4TH EUROPEAN ISOE WORKSHOP ON OCCUPATIONAL EXPOSURE MANAGEMENT AT NUCLEAR POWER PLANTS

### Introduction

The European ISOE Technical Centre co-organised with the International Atomic Energy Agency the Fourth European ISOE Workshop on Occupational Exposure at Nuclear Power Plants in March 2004, at Lyon, France. One hundred and ninety participants from 26 countries, European (all countries from western and central Europe with nuclear power plants) American (Canada and United States) and Asian (China, Japan, Korea) attended the meeting with a good balance between utilities, regulatory bodies and contractors. The IAEA supported participants from Central and Eastern European countries as well as from Eastern Asia. The workshop allowed 35 oral presentations and 28 posters presentations to be provided. A very informative exhibition was held by vendors and allowed participants to know more about their products during the coffee-breaks. All participants were split into small groups devoted to 10 pre-selected themes. Each group met twice and reached recommendations.

Five main recommendations were agreed on by the participants:

1. there is a need for harmonising regulations in order to maintain a high status of radiological protection at an international level in a deregulated context;
2. the regulatory bodies should also harmonise the contents of training, particularly in the context of workforce ageing;
3. the international organisations and regulatory bodies should take the lead to harmonise at the international level a dose passport for itinerant workers;
4. radiological protection indicators should be selected to help in optimising doses, provide indication for continuous improvement, estimate the effectiveness of radiological protection departments, provide means for benchmarking, create consistency between sites;
5. the radiological protection teams should increase their assistance "patrols" at workplaces.

For the first time, two specific meetings have preceded the workshop:

- one for the radiological protection managers from the NPPs; 30 participants attended that meeting;
- one for the senior representatives of the authorities; two topics were discussed there, the management of outside workers and the use of ISOE by the regulatory bodies.

## Conclusions and recommendations from the Workshop

A specific session on the radiological protection at the design stage of installations was mainly devoted to the new pressurised water reactor EPR. The Finish operator (TVO) and regulatory body (STUK) described their expectations in terms of occupational radiological protection. EDF, the French operator set-up a reasonable target of not exceeding 0.5 man.Sv per year (averaged for the life time of the reactor).

For the small group discussions, the participants as topics of interest particularly selected two topics:

- the setting up of radiological protection indicators (evaluation of the ALARA criteria);
- the needs in education and training in radiological protection.

The first was already one of the most selected topics at Portoroz (3rd European ISOE workshop in 2002) showing the growing importance of management of radiological protection and efficiency aspects in deregulated markets. The impact of deregulation was raised for the first time in 1998 at Malmö (first European ISOE Workshop)

The other topics for work in small groups were all selected by quite a number of participants and dealt with "Workers involvement and awareness", "Impact of deregulation", "Plant self assessment programmes", "Occupational exposure in case of emergency", "Management of itinerant workers" and "Loss of knowledge".

## The setting up of radiological protection indicators

In the context of competition, the setting up of goals and radiological protection indicators appears to the participants to be a very important management tool. These tools are more and more often used in the plants. Their goals must be measurable, realistic and challenging. They must be communicated to all stakeholders. The radiological protection specialists according to goals set up by the management may propose them. They should be then discussed with regulatory body. Deviations from the goals should require post job reviews.

> *Radiological protection indicators should be selected to help in optimising doses, provide indication for continuous improvement, estimate the effectiveness of radiological protecttion departments, provide means for benchmarking, create consistency between sites.*

## The needs in education and training in radiological protection

The participants pointed out the discrepancies between countries in terms of training both at initial and refreshing levels and the need for harmonisation. They stressed on the one hand the ageing of skilled workers as well as on the other hand the fact that many workers are well trained and committed to dose reduction. They also stressed the need for practical more than theoretical training, as experience appears often as important than training in achieving workers' involvement and awareness.

*The regulatory bodies should harmonise the contents of training.*

*The management should check and supervise regularly the implementation of training.*

The training courses should comprise practical exercises, use of mock up and awareness packages adapted to tasks and risks.

## Deregulation and radiological protection

The problem of the impact of deregulation on radiological protection was raised for the first time at Malmö in 1998 (first EC/ISOE Workshop). At that time, it appeared not to be a real concern. Two years later in 2000 at Tarragona (second European ISOE Workshop); the deregulation appeared clearly as a real challenge for the future for radiological protection. It led to a recommendation from the participants: "To consider new 'Radiation Protection' management techniques to avoid the potential negative impacts of deregulation on exposures, while keeping radiation protection independent from operation and maintenance of the plant".

The Lyon workshop confirmed what appeared for the first time at Portoroz, an "important reduction in radiological protection staff sizing, and loss of skilfulness". The present radiological protection specialists gave then warnings and recommendations:

- *The regulatory bodies should pay more attention to the deregulation process and negotiate with NPP's the minimum number of radiological protection and safety staff allowing to maintain a high status of radiological protection and ALARA.*
- *There is a need for harmonising regulations in order to maintain a high status of radiological protection at an international level in a deregulated context.*
- *The management should pay more attention to keep the quality of their contractors work through training and work management.*

## Other recommendations from the groups

The other working groups made recommendations endorsed by the participants to the workshop. It is therefore recommended that:

- *The international organisations and regulatory bodies take the lead to harmonise at the international level a **dose passport**;*
- *The radiological protection teams increase their **assistance "**patrols" at workplaces;*
- *Management and governments should recognise the **ageing of workforce** in NPPs and favour closer links with universities, personnel development plans, and adequate budget for long-term workforce replacement.*

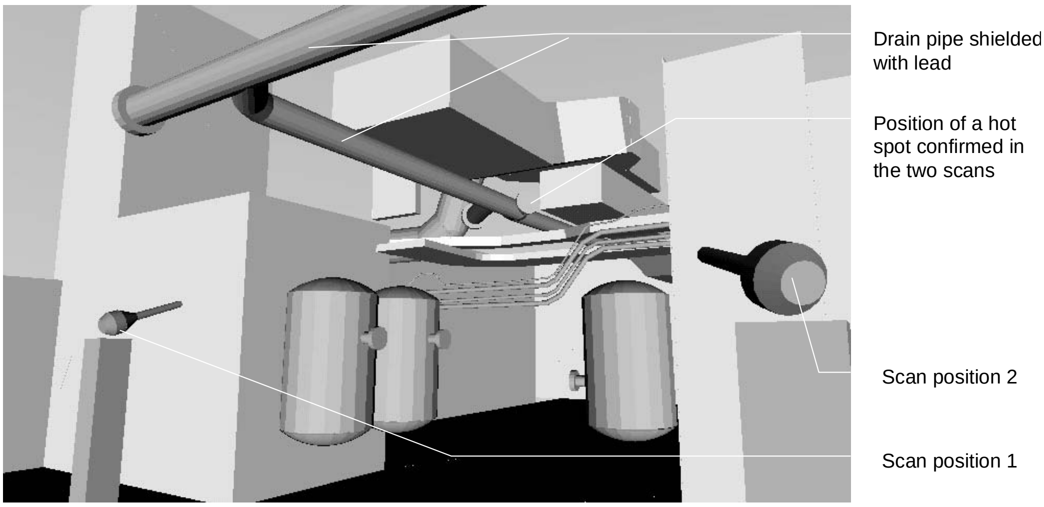
**Distinguished papers excerpts**

Three technical presentations were distinguished and invited to make their presentation in 2005 at the Miami ISOE North American international ALARA symposium in the United States of America. These papers were dealing with both technical and managerial problems and proposed very practical solutions.

***"Advantages of combining gamma scanning techniques and 3D dose simulation in dose optimisation problems"; F. Vermeersch, SCK•CEN Mol, Belgium***

This paper presents a method of combining results from gamma scanning equipment with a 3D dose simulation tool with the aim to achieving a reliable dose characterisation of the work site in order to perform dose assessment and optimisation for work planned in the area. The gamma scanning allows determining source positions, source geometry and sourcing composition. Hot spots that could be missed by traditional methods can be picked up. The combination of the gamma scanning and the radio-geometrical model makes it possible to perform a quantitative analysis of the source strengths leading to adapted dose assessments and optimisation.

Figure 1. **VISIPLAN model of the site geometry indicating the position of the gamma scanner during the two scans**




***"Recent International Developments on Contamination Limits on Packages B"; J. Hesse, RWE Power, Germany/B. Lorenz, GNS, Germany***

This paper presents the results of the IAEA Co-ordinated Research Project (CRP) on the Radiological Aspects of Package and Conveyance of Non-Fixed Contamination. One of the major tasks of the CRP has been from 2001 to 2003 to develop a new model for contamination limits for the Transport of radioactive material. This model to calculate doses from non-fixed surface is described in the presentation. It takes care of the types of radionuclides, the types of packages, the realistic description of tasks during a transport, the different exposure pathways and the possible exposures of workers and the public. The results of the model are presented in Bq/cm$^2$ corresponding for each nuclide to dose constraints of 2 mSv/year for the workers and 0.3 mSv/year for the public.

**Table 1. Final results (part of the complete table) for surface contamination levels in Bq/cm²**

| | Derived Level Workers [Bq/cm²] | | | | Derived Level Public [Bq/cm²] | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Nuclide | W-SM | W-SR | W-LR | W-FF | P-SM | P-SR | P-LR | P-FF | Overall Min. |
| Cm-248 | 0,1 | 1,6 | 2,1 | 1,7 | 31 | 6 | 68 | 46 | 0,1 |
| Co-55 | 135 | 134 | 305 | 407 | 1,8E+5 | 1,9E+4 | 4,0E+3 | 7,6E+3 | 134 |
| Co-56 | 81 | 80 | 185 | 249 | 9,7E+4 | 1,1E+4 | 2,3E+3 | 4,5E+3 | 80 |
| Co-57 | 2,3E+3 | 2,2E+3 | 4,9E+3 | 6,4E+3 | 2,3E+6 | 2,8E+5 | 6,8E+4 | 1,3E+5 | 2,2E+3 |
| Co-58 | 270 | 268 | 621 | 834 | 3,2E+5 | 3,6E+4 | 7,9E+3 | 1,5E+4 | 268 |
| Co-58m | 1,2E+5 | 1,3E+5 | 1,3E+5 | 1,3E+5 | 3,5E+8 | 7,1E+7 | 3,1E+8 | 3,5E+8 | 1,2E+5 |
| Co-60 | 109 | 108 | 245 | 323 | 1,1E+5 | 1,3E+4 | 3,2E+3 | 6,1E+3 | 108 |
| Cr-51 | 8,2E+3 | 8,2E+3 | 1,8E+4 | 2,4E+4 | 1,0E+7 | 1,1E+6 | 2,4E+5 | 4,7E+5 | 8,2E+3 |
| Cs-129 | 893 | 886 | 2,1E+3 | 2,8E+3 | 1,2E+6 | 1,2E+5 | 2,6E+4 | 4,9E+4 | 886 |
| Cs-131 | 7,7E+3 | 7,7E+3 | 1,7E+4 | 2,2E+4 | 1,0E+7 | 1,1E+6 | 2,4E+5 | 4,6E+5 | 7,7E+3 |
| Cs-132 | 362 | 359 | 836 | 1,1E+3 | 4,7E+5 | 5,0E+4 | 1,0E+4 | 2,0E+4 | 359 |
| Cs-134 | 129 | 128 | 227 | 266 | 1,7E+5 | 2,0E+4 | 4,9E+3 | 9,3E+3 | 128 |
| Cs-134m | 6,4E+3 | 6,4E+3 | 1,1E+4 | 1,2E+4 | 1,2E+7 | 1,2E+6 | 2,6E+5 | 5,0E+5 | 6,4E+3 |
| Cs-135 | 4,0E+3 | 4,6E+3 | 4,6E+3 | 4,6E+3 | 6,7E+6 | 1,4E+6 | 1,5E+7 | 1,0E+7 | 4,0E+3 |
| Cs-136 | 122 | 121 | 274 | 363 | 1,6E+5 | 1,7E+4 | 3,6E+3 | 6,9E+3 | 121 |
| Cs-137 | 284 | 283 | 439 | 487 | 3,8E+5 | 5,0E+4 | 1,3E+4 | 2,5E+4 | 283 |
| Cu-64 | 1,3E+3 | 1,3E+3 | 2,8E+3 | 3,6E+3 | 1,8E+6 | 1,9E+5 | 4,0E+4 | 7,6E+4 | 1,3E+3 |

"ALARA versus Reactor Safety concern – a practical case"; S. Hennigor, B. Ögren, Forsmark NPP, Sweden.

This presentation is a very practical one describing the modification of the moist separator (upper part of the steam dryer) at Forsmark BWR that took place in 2003 due to cracks. It describes the preparation of the work as well as its implementation and results (165.5 man mSv and maximum individual dose of 10.3 mSv). It points out that such type of work should be prepared at least one year ahead to collecting appropriate dose rate data, making a formal and comprehensive risk assessment and performing real optimisation with the contractor. It stressed the role of training on mock-up and the need of establishing follow-up meetings and radiological check-points with pre defined alternate actions.

## LIST OF PARTICIPANTS


### ARMENIA

| | | |
|---|---|---|
| Mr. Petros HOVHANNISYAN | Phone: | +374 1 2844232 |
| Armenian NPP | Fax: | +374 1 288580 |
| Medzamor | Email: | pamati@freenet.am |
| Armavir Region, 377766 | | |

### BELGIUM

| | | |
|---|---|---|
| Mr. Pascal DEBOODT | Phone: | +32 14 33 28 53 |
| SCK•CEN | Fax: | +32 14 32 16 24 |
| Service Interne de Prévention | Email: | pdeboodt@sckcen.be |
| et Protection au Travail | | |
| Boeretang 200 | | |
| B-2400 Mol | | |

| | | |
|---|---|---|
| Mr. Francis MOLITOR | Phone: | +32 2 289 21 75 |
| The Federal Agency for Nuclear Control | Fax: | +32 2 2892172 |
| (FANC) | Email: | francis.molitor@fanc.fgov.be |
| Rue Ravenstein 36 | | |
| B-1000 Bruxelles | | |

| | | |
|---|---|---|
| Mr. Fernand VERMEERSCH | Phone: | +32 14 33 27 11 |
| SCK•CEN | Fax: | +32 14 32 16 24 |
| Boeretang 200 | Email: | fvermeersch@sckcen.be |
| B-2400 Mol | | |

### BULGARIA

| | | |
|---|---|---|
| Mr. Georgi Grigorov VALTCHEV | Phone: | +359 973 72565 |
| Kozloduy NPP | Fax: | +359 973 80126 |
| 3321 Kozloduy | Email: | valtchev@npp.cit.bg |

### CANADA

| | | |
|---|---|---|
| Mr. Jean-Yves GAGNON | Phone: | +1 819 298 2943 Ext. 5165 |
| Hydro-Quebec | Fax: | +1 819 298 5660 |
| ALARA Co-Ordinator | Email: | gagnon.jean-yves@hydro.qc.ca |
| Centrale nucléaire Gentilly-2 | | |
| 4900, boul. Bécancour | | |
| Gentilly (Québec) G9H 13X3 | | |

Mr. Ricky KHALOO
Atomic Energy of Canada Ltd.
2251 Speakman Drive
Mississauga (Ontario) L5K 1B2

Phone: +905 823 9060 Ext: 4532
Fax: +905 822 0567
Email: khaloor@aecl.ca

**CHINA (PEOPLE' REPUBLIC OF)**

Mr. Haoyu WANG
DNMC CGNPC
Operation Dep. – Daya Bay NPP
Shenzhen 518124, Guangdong Province

Phone: +86 755 844 76 396
Fax: +86 755 844 76 466
Email: whyfree1977@yahoo.com.cn

Mr. Maochun YANG
DNMC, CGNPC
Operation Dep. – Daya Bay NPP
Shenzhen 518124, Guangdong Province

Phone. +86 755 844 79 486
Fax: +86 755 844 79 813
Email: yangmc@gnpjvc.com.cn

**CZECH REPUBLIC**

Ms. Bozena JUROCHOVA
Dukovany NPP
Personal Dosimetry Dept., CEZ, a.s.
675 50 Dukovany

Phone: +420 568 813 660
Fax: +420 568 866 360
Email: jurocb1.edu@mail.cez.cz

Mr. Stanislav KOCVARA
VF, a.s.
Namesti Miru 50
679 21 Cerna Hora

Phone: +420 516 428 611
Fax: +420 516 428 610
Email: skocvara@vf.cz

Mr. Petr OKRUHLICA
VF, a.s.
Namesti Miru 50
679 21 Cerna Hora

Email: pokruhlica@vf.cz

Ms. Karla PETROVA
State Office for Nuclear Safety
Senovázné náměstí 9
110 00 Prague 1

Phone: +420 2 2162 4556
Fax: +420 2 2162 4710
Email: karla.petrova@sujb.cz

**FINLAND**

Ms. Kirsi ALM-LYTZ
STUK
Radiation and Nuclear Safety Authority
PO Box 14
00881 Helsinki

Phone: +358 9 759
Fax: +358 9 759 88382
Email: kirsi.alm-lytz@stuk.fi

Mr Mickäel BÄCKSTRÖM
RADOS Technology Oy
P.O. Box 506, Mustionkatu 2
20101 Turku

Phone: +358 2 468 4660
Fax: +358 2 468 4601

Mr Erik LEHTONEN
RADOS Technology Oy
P.O. Box 506
Mustionkatu 2
20101 Turku
Phone: +358 2 468 4660
Fax: +358 2 468 4601

Mr. Timo HEISKANEN
Fortum Power and Heat Oy
Loviisa NPP
PO Box 23
07901 Loviisa
Phone: +358 10 45 55011
Fax: +358 10 45 54435
Email: timo.heiskanen@fortum.com

**Mr. Timo KONTIO**
Fortum Power and Heat Oy
Loviisa NPP
PO Box 23
07901 Loviisa
Phone: +358 10 45 55011
Fax: +358 10 45 54435
Email: timo.kontio@fortum.com

Ms. Suvi RISTONMAA
STUK
Radiation and Nuclear Safety Authority
PO Box 14
00881 Helsinki
Phone: +358 9 759 88442
Fax: +358 9 759 88382
Email: suvi.ristonmaa@stuk.fi

Mr. Jukka SOVIJÄRVI
TVO NPP, Teollisuuden Voima Oy
SF-27160 Olkiluoto
Phone: +358 2 8381 5510
Fax: +358 2 8381 5508
Email: jukka.sovijarvi@tvo.fi

**FRANCE**

Mr. Gonzague ABELA
EDF – GPR
Cap Ampère – Bâtiment Tête Pleyel
1, Place Pleyel
F-93282 Saint-Denis Cedex
Phone: +33 1 43 69 31 13
Fax: +33 1 43 69 13 73
Email: gonzague.abela@edf.fr

Mr. Jean-Baptiste ANDRIAMAHANINA
EDF – CNPE Cattenom
BP 41
F-57570 Cattenom
Phone: +33 3 82 51 71 18
Fax: +33 3 82 51 72 98
Email: donatien.colin@edf.fr

Mr. Jean-Pierre BERTHET
UNITECH
341 Avenue Thiers
F-33100 Bordeaux
Phone: +33 5 56 322 807
Fax: +33 5 56 409 337
Email: jean-pierre.brt@wanadoo.fr

Mr. Christian BESSON
NTC Nucléaire Service International
Z.I. de Gron
F-89100 Gron

Mr. Alain BLAIN
Département Radioprotection
Framatome ANP – SFXR, BAL 1323A
10 rue Juliette Récamier
F-69456 Lyon Cedex 06
Phone: +33 4 72 74 87 89
Fax: +33 4 72 74 70 29
Email: alain.blain@framatome-anp.com

Mr. Louis Patrick BARBE
EDF – CNPE Bugey
Service de Médecine du Travail, BP 14
F-01366 Camp-De-La-Valbonne Cedex
Phone: +33 4 74 34 32 48 OU 32 02
Fax : +33 4 74 34 18 17
Email: louis.barbe@edf.fr

Mr. Thierry BARRAS
EDF/BE-DIN Cap Ampère
1, Place Pleyel
F-93282 Saint-Denis Cedex
Phone: +33 1 43 69 04 19
Fax: +33 1 43 69 04 80
Email: thierry.barras@edf.fr

Mr. Stéphane CALPENA
DSNR Lyon
2, rue Antoine Charial
F-69003 Lyon
Phone: +33 4 37 91 43 88
Fax: +33 4 37 91 28 04
Email: stephane.calpena@asn.minefi.gouv.fr

Mr. Donatien COLIN
EDF – CNPE Cattenom
BP 41
F-57570 Cattenom
Phone: +33 3 82 51 71 14
Fax: +33 3 82 51 72 98
Email: donatien.colin@edf.fr

Mr. Philippe COLSON
EDF/DPN CAPE-GPR, Cap Ampère
1, Place Pleyel
F-93282 Saint-Denis Cedex
Phone: +33 1 43 69 31 07
Fax: +33 1 43 69 13 73
Email: philippe.colson@edf.fr

Mr. Olivier COUASNON
IRSN/DRPH/SER
BP 17
F-92262 Fontenay-Aux-Roses Cedex
Phone: +33 1 58 35 78 34
Fax: +33 1 58 35 88 29
Email: olivier.couasnon@irsn.fr

Mrs. Catherine COUTURIER
Framatome ANP
Direction des Services nucléaires
10, rue Juliette Récamier
F-69456 Lyon Cedex 06
Phone: +33 4 72 74 82 11
Fax: +33 4 72 74 70 29
Email:
catherine.couturier@framatome-anp.com

Mr. Pascal CROÜAIL
CEPN
BP 48 – Route du Panorama
F-92263 Fontenay-Aux-Roses Cedex
Phone: +33 1 58 35 74 60
Fax: +33 140 84 90 34
Email: lefaure@cepn.asso.fr

Ms. Lucie D'ASCENZO
CEPN
BP 48 – Route du Panorama
F-92263 Fontenay-Aux-Roses Cedex
Phone: +33 158 35 76 60
Fax: +33 1 40 84 90 34
Email: dascenzo@cepn.asso.fr

Mr. Hervé DELABRE
EDF – CNPE Cattenom
BP 41
F-57570 Cattenom
Phone: +33 3 82 51 77 02
Fax: +33 3 82 51 73 77
Email: herve.delabre@edf.fr

Mr. Philippe DENIMAL
Directeur des Services Nucléaires
Framatome ANP
10, rue Juliette Récamier
F-69456 Lyon Cedex 06
Email: philippe.denimal@framatome-anp.com

Mr. Bruno DESNOYERS
COGEMA Logistics
BP 302
F-78054 Saint-Quentin-en-Yvelines
Phone: +33 1 39 48 75 00 - 74 99
Fax: +33 1 39 48 75 01
Email: bdesnoyers@cogemalogistics.com

Mr. Philippe DESTENBERT
SynOdys Group
BP n° 1
F-13113 Lamanon
Phone: +33 4 90 59 60 00
Fax: +33 4 90 59 55 18
Email: vely@mgpi.com.fr

Mr. Olivier DIEUDONNE
SynOdys Group
BP n° 1
F-13113 Lamanon
Phone: +33 4 90 59 60 20
Fax: +33 4 90 59 60 89

Mr. Johannes ERNST
NTC Nucléaire Service International
Z.I. de Gron
F-89100 Gron

Mrs. Céline FASULO
DGSNR – SD2
10, Route du Panorama
F-92266 Fontenay-Aux-Roses Cedex
Phone: +33 1 43 19 71 04
Fax: +33 1 40 19 70 66
Email: c.fasula@asn.minefi.gouv.fr

Mr. le Docteur Jean-Louis FENOLLAND
EDF – CNPE de Golfech
BP 24
F-82401 Valence D'Agen Cedex
Phone: +33 5 63 29 38 09
Fax: +33 5 63 29 34 99
Email: jean-louis.fenolland@edf.fr

Mr. Roland FOREST
SynOdys Group
BP n° 1
F-13113 Lamanon
Phone: +33 4 90 59 60 34
Fax: +33 4 90 59 55 18

Mr. Matthias FRANZ
SynOdys Group
BP n° 1
F-13113 Lamanon
Phone: +33 4 90 59 60 08
Fax: +33 4 90 59 55 18

Mr. Yves GARCIER
EDF – Branches Energies
Division Production Nucléaire
Site Cap Ampère – Tête Pleyel B
1 place Pleyel
F-93282 Saint-Denis Cedex
Phone: +33 1 43 69 30 20
Fax: +33 1 43 69 34 95
Email: yves.garcier@edf.fr

Mrs. Christine GAURON
INRS
Dpt. Etudes et Assistance Médicale
30 rue Olivier Noyer
F-75680 Paris Cedex 14
Phone: +33 1 40 44 31 85
Fax: +33 140441415

Mr. Nicolas GIBERT
CNPE de Chinon
Service Prévention des Risques
BP 80
F-37420 Avoine
Phone: +33 2 47 98 90 65
Fax: +33 2 47 98 90 79
Email: nicolas.gibert@edf.fr

Mr. Sébastien GRENINGER
DGSNR – SD2
10, Route du Panorama
F-92266 Fontenay-Aux-Roses Cedex
Phone: +33 1 43 19 71 01
Fax: +33 1 40 19 70 66
Email: s.greninger@asn.minefi.gouv.fr

Mr. Olivier GUIMONNEAU
NTC Nucléaire Service International
Z.I. de Gron
F-89100 Gron

Mr. Luc GUINARD
EDF/SEPTEN/TE/RP
12-14 av. Dutriévoz
F-69628 Villeurbanne
Phone: +33 4 72 82 73 95
Fax: +33 4 72 82 73 56
Email: luc.guinard@edf.fr

Ms. Olvido GUZMAN LOPEZ-OCON
ASN – DGSNR – SD4
6, Place du Colonel Bourgoin
F-75572 Paris Cedex 12
Phone: +33 1 40 19 86 89
Fax: +33 1 40 19 87 90
Email: olvido.guzman-lopez@asn.minefi.gouv.fr

Mr. Christian HALLEPEE
CNPE Gravelines
BP 149
F-59820 Gravelines
Phone: +33 3 28 68 41 41

Mr. Didier HARANGER
EDF – DPN – FTC – GPR
Immeuble Cap Ampère « Tête Pleyel B »
1, Place Pleyel
F-93282 Saint-Denis Cedex
Phone: +33 1 43 69 31 47
Fax: +33 1 43 69 13 73
Email: didier.haranger@edf.fr

Mr. Claude HENRY
EDF – CNPE Bugey
BP 14
F-01366 Camp De La Valbonne Cedex
Phone: +33 4 74 34 30 01
Fax: +33 4 74 34 17 32
Email: claude.henryt@edf.fr

Mr. Laurent HIPPOCRATE
CNPE de Chinon
Service Prévention des Risques
BP 80,
F-37420 Avoine
Phone: +33 2 47 98 70 22
Fax: +33 2 47 98 90 79
Email: laurent.hippocrate@edf.fr

Mr. Thierry HOFFMANN
NTC Nucléaire Service International
Z.I. de Gron,
F-89100 Gron
Phone: +33 3 86 65 61 57
Fax: +33 3 86 95 53 22
Email: ntc@ntc-industrie.fr

Mr. Albert JAMANN
NTC Nucléaire Service International
Z.I. de Gron
F-89100 Gron

Mr. Bernard JEANNIN
CNPE du Blayais
BP 27
F-33820 Saint-Ciers-Sur-Gironde
Phone: +33 5 57 33 31 30
Fax: +33 5 57 33 32 89
Email: bernard.jeannin@edf.fr

Mr. Gérard JEANTON
EDF/UNIPE
Tour EDF Part Dieu
9, rue des Cuirassiers
BP 3181
F-69402 Lyon Cedex 03
Phone: +33 54 78 71 28 58
Fax: +33 4 78 71 29 00
Email: gerard.jeanton@edf.fr

Mr. André-Claude LACOSTE
Autorité de Sûreté Nucléaire
Direction Générale de la Sûreté Nucléaire
et de la Radioprotection
DIR-Bureau 2215
6, place du Colonel Bourgoin
F-75572 Paris Cedex 12
Phone: +33 1 40 19 86 17
Fax: +33 1 40 19 86 24
Email: andre-claude.lacoste@asn.minefi.gouv.fr

Mr. Francis LAFFERRIERE
CNPE du Blayais
BP 27
F-33820 Saint-Ciers-Sur-Gironde
Phone: +33 5 57 33 31 30
Fax: +33 5 57 33 32 89
Email: francis.lafferriere@edf.fr

Mr. Georges LANDRIN
CNPE St-Laurent-des-Eaux
BP 42
F-41220 Saint-Laurent-Nouan
Phone: +33 2 54 44 81 31
Fax: +33 2 54 44 85 11

Mrs. Audrey LANGLOIS
EDF/UTO/SIS
Immeuble « Maille Nord »
6, avenue Montaigne
F-93192 Noisy-Le-Grand Cedex
Phone: +33 1 49 32 76 86
Email: audrey.langlois@edf.fr

Mr. Bertrand LAUNAIS
SynOdys Group
BP n° 1
F-13113 Lamanon
Phone: +33 4 90 59 61 87
Fax: +33 4 90 59 55 18

Mr. Michel LAUNAY
EDF – CNPE de Belleville
BP 11
F-18240 Lere
Phone: +33 2 48 54 50 01
Fax: +33 2 48 54 08 38

Mr. Jacques LEBEAU
EDF/GPR/BE/DPN/CAPE
Immeuble Cap Ampère
Bât. Tête Pleyel
1 Place Pleyel
F-93282 Saint-Denis Cedex
Phone: +33 1 43 69 31 97
Fax: +33 1 43 69 13 73
Email: jacques.lebeau@edf.fr

Mr. Christian LEFAURE
CEPN, BP 48
Route du Panorama
F-92263 Fontenay-Aux-Roses Cedex
Phone: +33 1 58 35 79 08
Fax: +33 140 84 90 34
Email: lefaure@cepn.asso.fr

Mr. Dimitri LESECHE
CNPE St-Laurent-des-Eaux
BP 42
F-41220 Saint-Laurent-Nouan
Phone: +33 2 54 44 81 31
Fax: +33 2 54 44 85 11

Mr. Didier LHOMME
EDF – CNPE de Belleville
BP 11
F-18240 Lere
Phone: +33 2 48 54 52 26
Fax: +33 2 48 54 08 38
Email: didier.lhomme@edf.fr

Mr. Guy LIEUMONT
EDF/DPN
Projet Exploitation EPR
165 à 173 avenue Pierre Brossolette
BP 900
F-92542 Montrouge Cedex
Phone: +33 1 41 48 92 41
Fax: +33 1 41 48 90 09
Email: guy.lieumont@edf.fr

Mr. Michel LOUIS
EDF – CNPE Civaux
BP 64
F-86320 Civaux
Phone: +33 5 49 83 50 41
Fax: +33 5 49 83 58 89
Email: michel.louis@edf.fr

Mr. Gérald MACHICOANE
EDF/DPN/CAPE
GPR
Cap Ampère
1, Place Pleyel
F-93282 Saint-Denis Cedex
Phone: +33 1 43 69 15 30
Fax: +33 1 43 69 13 73
Email: gerald.machicoane@edf.fr

Mr. Eric MENNECIER
Framatome ANP
SFXR, BAL 1228A
10, rue Juliette Récamier
F-69456 Lyon Cedex 06
Phone: +33 4 72 74 81 07
Email: eric.mennecier@framatome-anp.com

Mr. Pascal MICHEL
Framatome ANP
10 rue Juliette Récamier
F-69456 Lyon Cedex 06
Phone: +33 4 72 74 87 87
Fax: +33 4 72 74 85 05
Email: pascal.michel@framatome-anp.com

Mr. Xavier MICHOUX
EDF/CNEN
165-173 Avenue Pierre Brossolette
BP 900
F-92542 Montrouge Cedex
Phone: +33 1 41 48 94 67
Fax: +33 1 41 48 92 65
Email: xavier.michoux@edf.fr

Mr. Dominique MINIERE
EDF – Site Cap Ampère
1, place Pleyel
F-93282 Saint-Denis Cedex
Phone: +33 1 43 69 47 91
Email: dominique.miniere@edf.fr

Mr. Charles PAURON
EDF – DPN/GPR
Cap Ampère – Bâtiment Tête Pleyel
1, Place Pleyel
F-93282 Saint-Denis Cedex
Phone: +33 1 43 69 12 17
Fax: +33 1 43 69 13 73
Email: charles.pauron@edf.fr

Ms. Marie-Claire PERRIN
EDF/SEPTEN
Département SN
12-14 avenue Dutriévoz
F-69628 Villeurbanne Cedex
Phone: +33 4 72 82 75 88
Fax: +33 4 72 82 76 90
Email: marie-claire.perrin@edf.fr

Mr. Samuel OZIL
Delta Protection
Z.A. de Berret
F-30200 Bagnols Sur Ceze
Phone: +33 4 66 89 18 36
Fax: +33 4 66 89 36 31
Email: sozil@bacou-dalloz.com

Mr. Steve PHILLIPS
SynOdys Group
BP n° 1
F-13113 Lamanon
Phone: +33 4 90 59 61 02
Fax: +33 4 90 59 55 18

Mr. Laurent POIRIER
NTC Nucléaire Service International
Z.I. de Gron,
F-89100 Gron

Mr. Don-Pierre POMPEI
COGEMA
2, rue Paul Dautier, BP 4
F-78141 Vélizy
Phone: +33 1 39 26 36 02
Fax: +33 1 39 26 27 40
Email: dpompei@cogema.fr

Mr. Brice PRUGNAUD
LCIE Landauer
33, av. de la Division Leclerc
F-92260 Fontenay-Aux-Roses
Phone: +33 1 40 95 62 95
Fax: +33 1 40 95 62 84
Email: brice.prugnaud@lcie.fr

Mr. Christophe QUINTIN
DSNR Lyon
2, rue Antoine Charial
F-69426 Lyon Cedex 03
Phone: +33 4 37 91 43 60
Fax: +33 4 37 91 28 00
Email: christophe.quintin@asn.minefi.gouv.fr

Mr. Alain QUIOT
EDF/UTO
Immeuble "Maille Nord"
6, avenue Montaigne
F-93192 Noisy-Le-Grand Cedex
Phone: +33 1 49 32 76 98
Fax: +33 1 49 32 79 19
Email: alain.quiot@edf.fr

Mr. Alexandre RIEDEL
EDF/DPN/CAPE
GPR
Site Cap Ampère
Tête Pleyel B
1, Place Pleyel
F-93282 Saint-Denis Cedex
Phone: +33 1 43 69 29 94
Fax: +33 1 43 69 13 73
Email: alexandre.riedel@edf.fr

Mr. Alain ROCHER
EDF – DPN
Cap Ampère
Bâtiment Tête Pleyel
1, Place Pleyel
F-93282 Saint-Denis Cedex
Phone: +33 1 43 69 31 56
Fax: +33 1 43 69 13 73
Email: alain.rocher@edf.fr

Mr. Sylvère ROGER
EDF – CNPE Paluel
BP 48
F-76450 Cany Barville
Phone: +33 2 35 57 62 80
Fax: +33 2 35 57 6259

Mr. Alain SALIOT
Delta Protection
Z.A. de Berret
F-30200 Bagnols Sur Ceze
Phone: +33 4 66 89 18 36
Fax: +33 4 66 89 36 31
Email: asaliot@bacou-dalloz.com

Mr. Timo SALOMAA
SynOdys Group
BP n° 1
F-13113 Lamanon
Phone: +33 4 90 59 66 39
Fax: +33 4 90 59 55 18
Email: vely@mgpi.com.fr

Mr. Patrice SAUMON
Framatome ANP
10, rue Juliette Récamier
F-69456 Lyon Cedex 06
Phone : +33 4 72 74 82 16
Fax: +33 4 72 74 82 49
Email: patrice.saumon@framatome-anp.com

Mr. Francis SCHULCZ
SynOdys Group
BP n° 1
F-13113 Lamanon
Phone: +33 4 90 59 60 15
Fax: +33 4 90 59 60 89

Mr. Bertrand SERISE
LCIE Landauer
33, avenue du Général Leclerc
F-92260 Fontenay-Aux-Roses

Mr. Jean-Bernard SERVAJEAN
EDF/GPR/BE/DPN/CAPE
Immeuble Cap Ampère
Bât. Tête Pleyel
1 Place Pleyel
F-93282 Saint-Denis Cedex
Phone: +33 1 43 69 28 02
Fax: +33 1 43 69 13 73
Email: jean-bernard.servajean@edf.fr

Ms. Margot TIRMARCHE
IRSN/DRPH/SRBE/LEPID
BP 17
F-92262 Fontenay-Aux-Roses Cedex
Phone: +33 1 58 35 71 94
Email: margot.tirmarche@irsn.fr

Mr. François VERGON
SynOdys Group
BP n° 1
F-13113 Lamanon
Phone: +33 4 90 59 61 04
Fax: +33 4 90 59 55 18

Ms. Françoise VIGNERIE
CNPE Gravelines
BP 149
F-59820 Gravelines
Phone: +33 3 28 68 45 33
Fax: +33 3 28 68 41 19
Email: francoise.vignerie@edf.fr

Mr. Dominique VINCENT
CNPE de Golfech, BP 24
F-82401 Valence D'Agen Cedex
Phone. +33 5 63 29 35 30
Fax: +33 5 63 29 34 53
Email: dominique-rene.vincent@edf.fr

Mr. Barthélémy VIVES
SynOdys Group
BP n° 1
F-13113 Lamanon
Phone: +33 4 90 59 60 30
Fax: +33 4 90 59 55 18
Email: vely@mgpi.com.fr

Mr. Philippe WEICKERT | Phone: +33 4 74 34 21 07
EDF/SPP | Fax: +33 4 74 34 21 18
Site du Bugey - Pôle Maintenance Thermique | Email: philippe.weickert@edfgdf.fr
BP 56
F-01360 Loyettes

Mr. Benoit ZERGER | Email: benoit.zerger@asn.minefi.gouv.fr
DSNR Lyon, 2, rue Antoine Charial
F-69003 Lyon

**Mr. Eric ZULESI** | Phone: +33 4 90 59 60 31
SynOdys Group | Fax: +33 4 90 59 55 18
BP n° 1
F-13113 Lamanon

**GERMANY**

Mrs. Dagmar DERDAU | Phone: +49 4152 15 25 80
Kernkraftwerk Krümmel GmbH | Fax: +49 4152 152099
Aktivitätsüberwachung | Email: d.derdau@kruemmel.de
Elbuferstrasse 82
D-21502 Geesthacht

Mr. Johannes HESSE | Phone: +49201 12 24878
RWE Power AG | Fax: +49 201 12 24107
Huyssenallee 2, PNN-R | Email: johannes.hesse@rwe.com
D-45128 Essen

Mr Peter JUNG | Phone: +49 7256 95 3287
EnBW Kraftwerke AG | Fax: +49 7256 95 3488
Kernkraftwerk Philippsburg (KKP) | Email: P.Jung@kkp.enbw.com
Postfach 1140
D-76652 Philippsburg

Mr. Heinz Peter KAPTEINAT | Phone: +49 201 81 28 248
VGB PowerTech e.V. | Fax: +49 201 81 28 345
CC Kernkraftwerke | Email: heinzpeter.kapteinat@vgb.org
Postfach 10 39 32
D-45039 Essen

Ms. Marlies LUSZIK-BHADRA | Phone: +49 531 592 6520
Physikalisch-Technische Bundesanstalt | Fax: +49 531 592 7205
Laboratory Neutron Dosimetry | Email : marlies.luszik-bhadra@ptb.de
Bundesallee 100
D-38116 Braunschweig

Mr. Wolfgang PFEFFER | Phone: +49 221 2068 773
Gesellschaft für Anlagen- | Fax: +49 221 2068 9902
und Reaktorsicherheit (GRS) mbH | Email: pff@grs.de
Schwertnergasse 1
D-50667 Köln

Mr. Josef SCHOBER
TÜV Süddeutschland Bau
und Betrieb GmbH
BB-ETS 3
Westendstrasse 199
D-80686 München
Phone: +49 89 5791 1562
Fax: +49 89 5791 1594
Email: josef.schober@tuev-sued.de

Mr. Ian Roger TERRY
Framatome ANP GmbH
Department NGPS4
Building 72
Freyeslebenstr. 1
D-91050 ERLANGEN
Phone: +49 9131 18 96336
Fax: +49 9131 18 97507
Email: Ian.Terry@framatome-anp.de

Mr. Thomas Van APPELDORN
E. On Kernkraftwerk GmbH
KW Wûrgassen
Zum Kernkraftwerk 25
D-37688 Beverungen
Phone: +49 5273 382160
Fax: +49 5273 382161
Email: thomas.appeldornvan@eon-energie.com

Mr. Daan VAN BREE
RADOS Technology GmbH
Ruhrstrasse 49
D-22761 Hamburg
Phone: +49 40 85193-0
Fax: +49 40 8593 256

Mr. Wolfgang WAHL
GSF – National Research Center
for Environment and Health
Institute of Radiation Protection
D-85764 Neuherberg
Phone: +49 89 3187 2348
Fax: +49 89 3187 3373
Email: wahl@gsf.de

**HUNGARIA**

Mr. Istavan VEGVARI
Hungarian Atomic Energy Authority
PO Box 676
1539 Budapest
Phone: +36 1436 4898
Fax: +36 1 436 4883
Email: vegvari@haea.gov.hu

**JAPAN**

Mr. Yasunori KOKUBUN
Tokyo Electric Power Company (TEPCO)
Nuclear Power Plant Management Dept.
Radiation Safety Control & Radiological
Emergency Mgt Gr.
1-1-3 Uchisaiwai-cho
1-chome
Chiyoda-Ku
Tokyo 100
Phone: + 81 3 4216 4819
Fax: + 81 3 3596 8547
Email: kokubun.yasunori@tepco.co.jp

Mr. Kazuhiro KOMORI
Japan Nuclear Energy Safety Org. (JNES)
Safety Inphoneligence Division
Fujitakanko Toranomon Bldg, 8th floor
17-1, 3-Chome Toranomon
Minato-Ku
Tokyo 105-0001
Phone: +81 3 4511 1941
Fax: +81 3 4511 1998
Email: komori-kazuhiro@jnes.go.

Mr. Masayuki KARUI
NEL
1-3-7 Tosa Buri,
Nishi-ku
Osaka 550-0001
Phone: +81 66 446 93 63
Fax: +81 66 446 1756
Email: makarui@sg-gr.neltd.co.jp

Ms. Akiko OGATA
Japan Nuclear Energy Safety Org. (JNES)
Safety Inphoneligence Division
Fujitakanko Toranomon Bldg, 8th floor
17-1, 3-Chome Toranomon
Minato-Ku
Tokyo 105-0001
Phone: +81 3 4511 1946
Fax: +81 3 4511 1998
Email: ogata-akiko@jnes.go.jp

**Mr. Miyazaki SHINICHIRO**
The Kansai Electric Power Co, Inc
3-3-22, Nakanoshima,
Kita-Ku
Osaka 530-8270
Phone: +81 6 7501 0151
Fax: +81 6 6441 4287-
Email: k576619@kepco.co.jp

**Mr. Masataka YAMADA**
NEL
1-3-7 Tosa Buri,
Nishi-ku
Osaka 550-0001
Phone. +81 66 446 93 63
Fax: +81 66 446 1756
Email: mayamada@sg-gr.neltd.co.jp

**KOREA**

Mr. Hee Jin AHN
Radiation Safety Section
YongGwang NPP Site – Division 2
514, Kyema-Ri
Hong Nong-Eup
YongGwang-Gun,
Chunnam, 513-880
Phone: +82 61 357 2285
Fax: +82 61 357 2219
Email: hjahn@khnp.co.kr

Mr. Dae Hyung CHO
Korea Institute of Nuclear Safety (KINS)
19 Guseong-dong
PO Box 114
Yuseong,
Daejon, 305-600
Phone: +82 42 868 0306
Fax: +82 42 868 0556
Email: k183cdh@kins.re.kr

Mr. Saeng-ki KIM
Nuclear Safety & Technology Dept. KHNP
167, Samsung-dong,
Kangnam-gu
Seoul 135-791
Phone: +82 2 3456 2430
Fax: +82 2 3456 2429
Email: kimsk3@khnp.co.kr

Mr. Seong Ho NA
Korea Institute of Nuclear Safety (KINS)
19 Guseong-dong
PO Box 114
Yuseong
Daejon 305-600
Phone: +82 42 868 0302 ou 0036
Fax: +82 42 862 3680
Email: shna@kins.re.kr

**LITHUANIA**

Mr. Kestutis GEDIMINSKAS
Ignalina NPP
LT-4761 Visaginas
Phone: +370 66 28 354
Fax: +370 66 611 71
Email: gedimin@mail.inl.lt

Mr. Gintautas KLEVINSKAS
Radiation Protection Centre
Kalvariju 153
LT-2042 Vilnius
Phone: +370 5 2644720
Fax: +370 5 2644721
Email: g.klevinskas@rsc.lt

**LUXEMBOURG**

Mr. Klaus SCHNUER
Commission Européenne
DG Energie et Transports,
Radioprotection H4
Centre A. Wagner C331, Rue A. de Gasperi
Plateau de Kirchberg
L-2920 Luxembourg Grand Duche
Phone: +352 4301 3 6388
Fax. +352 4301 3 4646
Email: klaus.schnuer@cec.eu.int

**NETHERLANDS**

Mr Frans KAMPING
N.V. EPZ, BORSSELE NPP
Lokatie Zeeland
Wilhelminahofweg 3
4454 PM Borssele, Postbus 130
NL-4380 AC Vlissingen
Phone: +31 113 35 6359
Fax: +31 113 35 2550
Email: F.kamping@epz.nl

**ROUMANIA**

Mrs. Catalina CHITU
CNE-Prod Cernavoda NPP
Strada Medgidiei Nr.1
PO Box 42
8625 Cernavoda
Phone: +402 41 239340
Fax: +402 41 239269
Email: cchitu@cne.ro

**SLOVAKIA**

Mr. Jan KAIZER
Bohunice NPP
919 31 Jaslovske Bohunice
Phone: +42 1 33 597 3673
Fax: +42 1 33 597 4786
Email: kaizer_jan@ebo.seas.sk

Mr. Lubomir DOBIS
Bohunice NPP
Radiation Protection Department
Atomové Elektrárne
919 31 Jaslovske Bohunice
Phone: +421 33 597 2128
Fax: +421 33 597 4786
Email: dobis_lubomir@ebo.seas.sk

Ms. Mariana MANCIKOVA
Mochovce NPP
Radiation Protection Dept.
935 39 Mochovce
Phone: +421 366363560
Fax: +421 36 6391159
Email: mancikova.mariana@emo.seas.sk

Mr. Jozef SÜSS
Mochovce NPP
Radiation Protection Department
935 39 Mochovce
Phone: +421 36636663235
Fax: +421 366391120
Email: suss_jozef@emo.seas.sk

Mr. Milan ZRUBEC
Mochovce NPP
Radiation Protection Dept.
935 39 Mochovce
Phone: +421 366 363932
Fax: +421 36603910120
Email: zrubec.milan@emo.seas.sk

Mr. Jaroslav SVITEK
Bohunice NPP
Atomové Elektrárne
919 31 Jaslovské Bohunice
Phone: +421 33 597 3298
Fax: +421 33 597 4786
Email: svitek_jaroslav@ebo.seas.sk

Mr. Dusan VIKTORY
Statny Zdravotny Ustav SR
Centrum Ochrany zdravia pred ziarenim
Trnavska 52,
826 45 Bratislava
Phone: +421 2 443 72287
Fax: +421 2 443 72619
Email: viktory@szusr.sk

**SLOVENIA**

Ms. Helena JANZEKOVIC
Slovenian Nuclear Safety Administration
Zelezna cesta 16, PO Box 5759
1001 Ljubljana
Phone: +386 1472 1194
Fax: +386 1472 1199
Email: helena.janzekovic@gov.si

**SPAIN**

Mr. Ramiro FRAGIO RODRIGUEZ
Central Nuclear de Cofrentes
46625 Cofrentes (Valencia)
Phone: +34 96 189 4300
Fax: +34 96 219 6477
Email: rjfragio@iberdrola.es

Mr. Juan José MONTESINOS
Consejo de Seguridad Nuclear
Area de Protección Radiológica de los Trabajadores
Justo Dorado n°11
28040 Madrid
Phone: +34 91 346 06 34
Fax: +34 91 346 05 88
Email: jjmc@csn.es

Ms. Maria-Luisa ROSALES
Consejo de Seguridad Nuclear
Justo Dorado, 11
28040 Madrid
Phone: +34 91 346 0128
Fax: +34 91 346 05 88
E-mail: mlrc@csn.es

Mr. Domingo SUSTACHA DUO
Central Nuclear de Almaraz
Apartado de Correos 73
10300 Navalmoral de la Mata (Caceres)
Phone: +34 92 702 52 32
Fax: +34 92 754 50 90 -
Email: d.sustacha@cnat.es ore dsd@cnat.es

Mrs. Isabel VILLANUEVA DELGADO
Consejo de Seguridad Nuclear
Justo Dorado, 11
28040 Madrid
Phone: +34 91 346 06 37
Fax: +34 91 346 05 88
E-mail: ivd@csn.es

**SWEDEN**

Mr. Paul BERGERSTAM
OKG Aktiebolag
S-572 83 Oskarshamn
Phone: +46 491 786 168
Fax: +46 491 787 661
Email: paul.bergerstam@okg.sydkraft.se

Mr. Magnus BENGTSSON
Ringhals Nuclear Power Plant
S-430 22 Väröbacka
Phone: +46 340 6672 51
Fax: +46 340 6680 00
Email: magnus.b.bengtsson@ringhals.se

Mr. Jan-Ove ERIKSSON
OKG Aktiebolag
S-572 83 Oskarshamn
Phone: +46 491 786 365
Fax: +46 491 786 958
Email: jan-ove.eriksson@okg.sydkraft.se

Mr. Staffan HENNIGOR
Forsmark NPP, Forsmarks Kraftgrupp AB
S-742 03 Östhammar
Phone: +46 17382076
Fax: +46 173 81850
Email: sig@forsmark.vattenfall.se

Mr. Jonas KARLSSON
Ringhals AB
Ringhals 1 NPP
Dep. R1MS
S-430 22 Väröbacka
Phone: +46 340 66 72 61
Fax: +46 340 6680 00 -
Email: jonas.karlsson@ringhals.se

Mr. Kjell LINDBERG
Forsmark NPP
Forsmarks Kraftgrupp AB
S-742 03 Östhammar
Phone: +46 173 81505
Fax: +46 173 82417
Email: krl@forsmark.vattenfall.se

Mr. Carl Göran LINDVALL
Barsebäck Kraft AB
Barsebäck Nuclear Power Plant
Box 524
S-246 25 Löddeköpinge
Phone: +46 4672 4350
Fax: +46 46 724 580
Email: carl-goran.lindvall@barsebackkraft.se

Mr. Ingemar LUND
Swedish Radiation Protection Authority (SSI)
S-171 16 Stockholm
Phone: + 46 8 729 7157
Fax: +46 8 729 7108
Email: ingemar.lund@ssi.se

Mrs. Hanna ÖLANDER-GÜR
Swedish Radiation Protection Authority (SSI)
S-17116 Stockholm
Phone: +46 87297100
Fax: +46 87297108
Email: hanna.olander.gur@ssi.se

Mr. Christer SOLSTRAND
OKG Aktiebolag
S-572 83 Oskarshamn
Phone: +46 491 786 406
Fax: +46 491 78 7602
Email: christer.solstrand@okg.sydkraft.se

Mr. Torgny SVEDBERG
Ringhals NPP
Dept. PRRTTU Technical Analysis
S-430 22 Väröbacka
Phone: +46 340 66 72 60
Fax: +46 340 66 51 02
Email: torgny.svedberg@ringhals.se

**SWITZERLAND**

Mr. Beat BITTERLI
NPP Gösgen
Kernkraftwerk Gösgen Daeniken AG
Postfach
CH-4658 Daeniken
Phone: +41 62 22220 00
Fax: +41 62 2 28 2001
Email: bbitterli@kkg.ch

Mr. Loys GINDRAUX
COMET AG
Business Unit Dosimetry
Herrengasse 10
CH-3175 Flamatt
Phone: +41 31 744 9298
Email: loys.gindraux@comet.ch

Mr. Swen Gunnar JAHN
Swiss Federal Nuclear Safety Insp. HSK
Sektion Radiologischer Arbeitsschutz
Koordination Nuklearer Forschungsanlagen
CH-5232 Villigen – HSK
Phone: +41 56 310 3931
Fax: +41 56 310 4941
Email: jahn@hsk.ch

**UKRAINE**

Ms. Tetyana LISOVA
Ministry of Fuel and Energy of Ukraine
State Nuclear Energy Department
Radiation Safety Division
30-34 Khreshchatyk str.
01601 Kyiv-1
Phone: +380 44 228 6563
Fax: +380 44 228 6563
Email: lisova@mintop.energy.gov.ua

**UNITED KINGDOM**

Mr. Matthew LUNN
British Energy Generation Ltd.
Environmental Support Section
Sizewell B Power Station
NR Leiston
Suffolk IP16 4UR
Phone: +44 1728 653750
Fax: +44 1728 653 189
Email: matthew.lunn@british-energy.com

Mr. Anastasios M. ZODIATES
British Energy Generation Ltd
Radiological Protection,
Barnett Way, Barnwood
Gloucester GL4 3RS
Phone: +44 1452 653 915
Fax: +44 1 452 652 206
Email: tasos.zodiates@british-energy.com

**UNITED STATES**

Mr. George BAKEVICH
UNITECH
295, Parker Street
PO Box 51957
Springfield, MA 01151
Phone: +1 413 543 6911x17
Fax: +1 413 543 2975
Email: george@u1st.com

Mr. Gunter BRUCKNER
UNITECH
3312 Preakness Drive
Flower Mound, TX 75028
Phone: +1 972 699 3312
Fax: +1 972 899 3313
Email: gunter@u1st.com

Mr. Frank M. GAVILA
F&J Specialty Products, Inc.
404 Cypress Road
PO Box 2888
Ocala, Florida 34478-2888
Phone: +1 352 680 1177
Fax: +1 352 680 1454
Email: fandj@fjspecialty.com ou
fmgavila@hotmail.com

Mr. Eric M. GOLDIN
San Onofre Nuclear Generating Station
PO Box 128
Mail Stop D3D
San Clemente, CA 92674-0128
Phone: +1 949 368 7532
Fax: +1 949 368 7575
Email: goldinem@songs.sce.com

Mr. Gregg JOHNSTONE
UNITECH
295, Parker Street, PO Box 51957
Springfield, MA 01151
Phone: +1 413 543 6911x12
Fax: +1 413 543 2975
Email: gregg@u1st.com

Mr. David W. KOZIN
Chemistry Department
DC Cook NPP, American Electric Power
One Cook Place
49106 Bridgeman, MI.
Phone: +1 269 465 5901
Fax: +1 269 466 2550
Email: DWKozin@aep.com

| | |
|---|---|
| Mr. Sergio LOPEZ<br>MGP Instruments Inc – SynOdys Group<br>5000 Highlands Parkway, Suite 150<br>Smyrna , Georgia 300082 | Phone: +1 770 432 27 44<br>Fax: +1 770 432 91 79 |
| Mr. David MILLER<br>ISOE North American Technical Center<br>College of Engineering MC-234<br>206 Nuclear Eng. Lab, University of Illinois<br>103 S. Goodwin Ave.<br>Urbana, Illinois 61801-2984 | Phone: +1 217 333 1098<br>Fax: +1 217 333-2906<br>Email: DWMPHD@aol.com |
| Mr. Kenneth OHR<br>Quad Cities NPP<br>Exelon Corporation<br>22710 206th Avenue North<br>Cordova, IL 91242 | Phone: +1 309 227 2725<br>Fax: +1 309 227 2265<br>Email: kenneth.ohr@exeloncorp.com |
| Mr. Eric OLSON<br>S.M. Stoller Corporation<br>990 S. Public Rd., Suite A<br>Lafayette, CO 80026 | Phone: +1 303 545 4410<br>Fax: +1 303 443 1408<br>Email: eolson@stoller.com |
| Mrs. Patricia ROBINSON<br>(n.p.) ENERGY Inc.<br>4526 Red Fox Rd.<br>Fort Collins, CO 80526 | Phone: +970 377 1444<br>Fax: +970 377 0105<br>Email: npeincl@aol.com |
| Mrs. Audrey SUMMERS<br>MGP Instruments Inc – SynOdys Group<br>5000 Highlands Parkway, Suite 150<br>Smyrna , Georgia 300082 | Phone: +1 770 432 27 44<br>Fax: +1 770 432 91 79 |
| Mr. Mike WILSON<br>MGP Instruments Inc – SynOdys Group<br>5000 Highlands Parkway, Suite 150<br>Smyrna , Georgia 300082 | Phone: +1 770 432 27 44<br>Fax: +1 770 432 91 79 |

**INTERNATIONAL ORGANISATIONS**

| | |
|---|---|
| Mrs. Monica GUSTAFSSON<br>IAEA, Div. of Radiation and Waste Safety<br>PO Box 100, Wagramerstr. 5<br>A-1400 Vienna | Phone: +43 1 2600/ 22725<br>Fax : +431 26007<br>Email: M.Gustafsson@iaea.org |
| Mr. Khammar MRABIT<br>IAEA, Div. of Radiation and Waste Safety<br>Dept. of Nuclear Safety<br>Wagramerstr. 5, PO Box 100<br>A-1400 Vienna | Phone: +43 1 2600 22722<br>Fax: +43 1 26007<br>Email: K.Mrabit@iaea.org |

| | | |
|---|---|---|
| Mr. Stefan MUNDIGL | Phone: | +33 1 45 24 10 45 |
| Mr. Brian Ahier | Fax: | +33 1 45 24 11 10 |
| OECD Nuclear Energy Agency | Email: | brian.ahier@oecd.org |
| Radiation Protection and | | |
| Radioactive Waste Management Division | | |
| Le Seine St. Germain | | |
| 12, Bld des Iles | | |
| F-92130 Issy-Les-Moulineaux | | |

| | | |
|---|---|---|
| Ms. Solange QUARMEAU | Phone: | +33 1 45 24 10 13 |
| OECD Nuclear Energy Agency | Fax: | +33 1 45 24 11 10 |
| Le Seine St. Germain | Email: | solange.quarmeau@oecd.org |
| 12, Bld des Iles | | |
| F-92130 Issy-Les-Moulineaux | | |

OECD PUBLICATIONS, 2 rue André-Pascal, 75775 PARIS CEDEX 16
Printed in France.

OECD PUBLICATIONS, 2, rue André-Pascal, 75775 PARIS CEDEX 16
PRINTED IN FRANCE
(66 2005 07 1) – No. 54417 2005